4. Advanced Nuclear Energy Systems

This program element includes 47 research projects to date of which 20 were awarded in FY 1999, 8 in FY 2000, 7 in FY 2001, and 12 in FY 2002. It includes the investigation and preliminary development of advanced concepts for reactor and power conversion systems. These systems offer the prospect of improved performance and operation, design simplification, enhanced safety, and reduced overall cost. Projects may involve innovative reactors; system or component designs; alternative power conversion cycles for terrestrial applications; new research in advanced digital instrumentation and control and automation technologies; hydrogen production from nuclear reactors; and identification and evaluation of alternative methods, analyses, and technologies to reduce the costs of constructing future nuclear power plants.

Additionally, this element includes research projects to improve the intrinsic proliferation-resistant qualities of advanced reactors and fuel systems. Possible technology opportunities and subjects of investigation include alternative proliferation-resistant reactor concepts, systems that minimize the generation of weapons-usable nuclear materials (e.g., Pu-239) and waste by-products, or systems that increase energy extraction from the utilization of plutonium and other actinide isotopes generated in the fuel.

Projects involving advanced reactors under this program element specifically address, among other items, the characteristics, feasibility, safety features, proliferation-resistance, and economic competitiveness of reactor systems, and additional research that may be required. These reactor concepts include advancements in light water reactor technology to achieve higher performance, or development of other higher temperature advanced reactor designs for higher efficiencies.

Other advanced-reactor concepts include compact or modular reactor designs suitable for transport to remote locations, and alternative energy production or co-generation reactor applications. Desirable features include long-lived reactor cores that minimize or avoid altogether the need for refueling, and concepts that maximize fuel burn-up or employ advanced energy conversion technology.

Finally, this program element includes research and development to identify and evaluate new and innovative concepts for producing hydrogen using nuclear reactors. This research includes investigation of hydrogen generation processes compatible with advanced reactor systems, and the integrating parameters needed to develop systems that are efficient and cost-effective overall.

Nuclear Energy Research Initiative

Directory Advanced Nuclear Energy Systems Project Summaries and Abstracts

NUCLEAR ENERGY RESEARCH INITIATIVE

Application of Innovative Experimental and Numerical Techniques for the Assessment of Reactor Pressure Vessel Structural Integrity

Primary Investigator: T.Y. Chu, Sandia National Laboratories (SNL)

Collaborators: U.S. Nuclear Regulatory Commission (NRC); Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA)

Project Number: 99-018

Project Start Date: August 1999

Project End Date: March 2003

Research Objectives

The Nuclear Energy Research Initiative (NERI)/NRC/OECD sponsored program consists of eight international partners: Belgium, Czech Republic, Finland, France, Germany, Spain, Sweden, and the United States. U.S. support is provided by the NRC and the Department of Energy's NERI program. This experimental/analytical program builds on the accomplishments of a previous NRC-sponsored Lower Head Failure (LHF) program (NUREG/CR-5582). The current program is referred to as the OECD Lower Head Failure (OLHF) program to distinguish it from the previous program and to recognize the international participation of the OECD.

This project consists of both experimental and analytical efforts in investigating the structural integrity of reactor pressure vessels. Experiments simulating the thermal/mechanical loads to a reactor pressure vessel generate data that can be implemented into a finite element code, such as the commercially available code ABAQUS, to assess the ability of the code to capture the response of the pressure vessel to severe accident conditions. In addition, the pressure vessel material (SA533B1 steel) used in these experiments is prototypical of reactor pressurized water reactor vessel material and is well-characterized by material property testing as part of this program.

The lower head of the reactor pressure vessel (RPV) can be subjected to significant thermal and pressure loads in the event of a core meltdown accident. The mechanical behavior of the reactor vessel's lower head is of importance both in severe accident assessment and the assessment of accident mitigation strategies. For severe-accident assessment, the failure of the lower head defines the initial conditions for all ex-vessel events, and in accident mitigation the knowledge of mechanical behavior of the reactor vessel defines the possible operational envelope for accident mitigation. The need for validated models of the lower head in accident scenarios is accomplished by well-controlled, well-characterized, large-scale experiments simulating realistic thermal/mechanical loads to the reactor pressure vessel. The purpose of the OLHF project is to investigate lower head failure for conditions of low reactor coolant system (RCS) pressure (2-5 MPa) and prototypic temperature differences across the vessel wall (ΔT_W) of 200°K to 400°K.

Research Progress

The previous NRC-sponsored research investigated the condition of high RCS pressures and small ΔT_W. Low RCS pressure is chosen because of the desire to use the data to develop models for assessing accident management strategies involving RPV depressurization. Pressure transient data is useful in assessing the effect of water injection as part of an accident management strategy. Prototypic ΔT_W is important because of the important need to provide data where stress redistribution occurs in the vessel wall (as a result of decreasing material strength with temperature). The OLHF experiments have been performed using 1 to 4.85 linear scale models of a typical pressurized water reactor (PWR) lower head. Figure 1 is a picture of the OLHF test assembly. The test vessel is geometrically scaled to a typical PWR reactor to preserve attributes of the membrane stress.

The test vessel consists of a 91.4 cm inner diameter, nominally 70 mm thick SA533B1 steel hemisphere welded to a 45 cm upright cylinder assembly closed off on top by a blank flange. The vessel is heated from within with a unique induction-heated graphite-radiating cavity. Large ΔT_W is achieved by increasing (with respect to geometrical scaling) the wall thickness and leaving the wall non-insulated. The membrane stress is preserved by increasing the test pressure by a factor corresponding to the wall thickness distortion (RW), i.e., $P_{test}=RW\bullet P_{RCS}$. The prototypic material for U.S. PWRs, SA533B1, was used to preserve material behavior.

Four integral tests were performed as part of the OLHF project. OLHF-1 and OLHF-2 are performed at 2 MPa and 5 MPa RCS pressure respectively. OLHF-3 examined the effect of pressure transients as the vessel wall passed through the ferrite-austenite phase transition region while the RCS pressure increased from an initial pressure of 2 MPa to a transient upper plateau of 5 MPa. The final test, OLHF-4, examined the effect of penetrations on vessel failure and was also tested at 2 MPa RCS (similar to OLHF-2). The bottom 120° of the test vessel was uniformly heated with a heat-up rate of 12°K/min for all tests.

Figure 1. The photograph is a post-test view of the OLHF-2 vessel on the test pad.

Two extra vessels were fabricated from the SA533B1 steel from which samples were prepared for material property testing. This material underwent essentially the same heat treatment and work history as actual test vessels to minimize variability in material properties. Five tensile tests were performed at high temperatures (925°K to 1,275°K) and 21 creep tests were performed at about 75 percent and 95 percent of yield stress at high temperatures (925°K to 1,275°K). Two replicate creep tests were performed to provide a means of assessing the reproducibility of test results. Supplemental tests were also performed by the French Atomic Energy Commission (CEA[1]) of France to extend and/or verify the database. The results of these tests are used to construct a constitutive model for implementation into structural analysis models.

Independent numerical simulations were performed for the OLHF-1 test by all OECD partners in an international benchmark activity. Using finite element methods or analytical calculations together with material property data and test data for geometry description and boundary conditions, participants evaluated the time to failure or the value of damage parameters at the experimental failure time as well as variations with time of several mechanical variables. Overall, numerical models correctly estimate the timing, mode, and location of vessel failure. However, more work is needed in developing models of the crack opening and crack propagation to assess the size of the breach.

Simple "engineering" methodologies are required in severe accident codes (e.g., MELCOR, SCDAP/RELAP5, and MAAP4) that model the full sequence of events that occur in a core melt accident. It has been demonstrated (NUREG/CR-5582) that the creep-based methods utilizing the semi-empirical lifetime rule nominally correlate both the time for onset of creep and failure times observed in the NRC-sponsored LHF tests. These models have been assessed against the experimental data where large through-wall temperature gradients are important, resulting in significant redistribution of internal stress from the hot inside surface to the cooler outside surface of the lower head.

Summary of Project Accomplishments

A summary is provided for the entire project. The accomplishments will be presented in terms of the three key program elements: integral experiments, material characterization, and model development and validation. Material characterization is the link between integral experiments and modeling.

Key observations from integral experiments

(1.) Large temperature differential leads to failure at higher inside wall temperature.

Comparison of the results of OLHF-1 and earlier tests (i.e., LHF-7) demonstrates the importance of stress redistribution on vessel deformation. Both tests performed with large through-wall temperature differences (OLHF-1 and OLHF-2) showed signs of non-linear deformation at higher temperatures than similar tests with small through-wall temperature differentials (LHF). The LHF tests showed signs of non-linear deformation

[1] Commissariat à L'Energie Atomique

when the inside surface was well below the yield stress. For the conditions of the OLHF tests, the onset of nonlinear deformation occurs as more than 10 percent of the vessel wall exceeds the yield stress. Failure in the OLHF tests occurred at higher temperatures than in corresponding LHF tests (small temperature differential). Since penetration failure was governed by global vessel deformation, this was also true of the penetration test.

(2.) Failures are typically localized.

Both the LHF and OLHF experiments as well as failure of reactor vessel retention (FOREVER) experiments revealed that the initiation of the failures is typically local. Failure was found to initiate at the location of maximum membrane to yield stress ratio. For OLHF experiments, since the load is carried by the cooler outer region of the wall, the stress ratio is evaluated at the external wall temperature. For the case of uniform temperature distribution, the crack initiates in the thinnest region because the location corresponds to maximum membrane stress. The crack initiates at the highest temperature regions for the case of non-uniform temperature distribution because yield stress is minimum at the highest temperature region.

Following this, most of the tests exhibited a localized propagation. Nevertheless, re-pressurization at elevated temperature can lead to rapid onset of non-linear deformation and failure and result in larger failure sites. This was observed in both OLHF-3 and LHF-5 tests.

It is important to note that the test results cannot be directly used to assess failure size for reactor cases because the rate of depressurization depends on the gas volume in the system, which is not scaled in the OLHF or the LHF tests. Furthermore, in the reactor case, molten corium ejection could enlarge the failure site through ablation.

(3.) Consistent global failure strain was observed.

The critical effective strain is often used as a failure criterion in modeling the vessel deformation in Severe Accident Codes. The critical effective failure strain for uniformly heated vessel without penetration was found to be ~30 percent. This value is consistent among the first three OLHF tests, and uniformly heated LHF tests.

Penetration failure has been found to occur as a result of global deformation of the lower head leading to failure at the weld vessel interface. Penetration failure occurs at much lower effective strain, i.e., 10 percent for OLHF-4 and 7 percent for LHF-4. In both experiments, failure initiated at the weld-vessel interface, which resulted in depressurization and termination of experiments. Direct application of the results to the reactor cases is questionable because understanding of the scaling effect of weld failure is currently lacking; furthermore, the effects of molten corium is not simulated in the OLHF experiments. Provisionally, one can assume that initiation of penetration failure due to global deformation occurs at a critical effective strain of ~10 percent. Metallurgical and material properties of weld are important topics that should be addressed for developing predictive model of penetration failure.

Key observations from material characterization experiments

The tensile and creep properties were measured for temperature up to approximately 1,300°K for LHF and OLHF steel. There is general consistency between the data from SNL and CEA and the external database established in the LHF program. The data fits (except elastic modulus) developed in the LHF program were used in OLHF analyses and numerical simulations.

However, a close examination of the data indicates that for future analysis of OLHF integral experiments it might be advisable to refit the data with more weight given to measured OLHF material properties. Fits giving equal weights to all data can be used for general assessment purposes. It is also important to note that comparisons between the SNL and CEA measurements indicate that there are laboratory-to-laboratory differences in the OLHF material data set. The effort and resources required to resolve such difference are likely to be quite high. At the moment it is perhaps best to acknowledge the difference as an irreducible uncertainty.

CEA has performed metallurgical characterization of failure sites of LHF and OLHF material. For high temperatures (> 1000°K), LHF material exhibits brittle failure whereas OLHF material exhibits ductile behavior.

CEA concluded that the difference could be attributed to the nearly 10 fold higher sulfur content in the LHF material as compared to the OLHF material although both materials conform to the specifications of SA533B1 steel. The effect of the ductile versus brittle behavior on critical strain, failure propagation, and final failure size should be assessed.

The ferrite to austenite phase transition region for the OLHF material has been determined to be between 1005°K and 1133°K using dilatometry. The tests also showed that there are no orientation effects although the grain structure does indicate the effect of cold work, i.e., finer grain near the inner and outer surfaces. The rate of vessel deformation diminished or reversed as the vessel wall passed through the phase transition from ferritic to austenitic steel. However, the effect of phase transition appears not to be as significant as once considered probably because only a portion of the wall is in the transition region at any instance of time and the induced stresses due to phase transition are of second order.

Key Observations on Severe Accident Codes and Numerical Simulations in Model Development and Validation

The OLHF project has produced a unique and well-qualified data set for code validation.

A benchmark exercise based on the OLHF-1 test data has been conducted by the OLHF participants. Different failure criteria have been used (damage or strain) by the participants with no consensus or convincing argument for any preference. Generally, the predicted failure times calculated by participants agree reasonably well with the test data. All the models, irrespective of their complexity, gave reasonable prediction of the failure time. It should be noted that failure time is not a rigorous measure of model predictability because the time to non-linear deformation is long compared to the time interval between the initiation of non-linear deformation and vessel failure. When referenced to the onset of non-linear deformation, there is still considerable spread in the predicted time to failure.

The OLHF-4 (penetration) vessel provides a snapshot of the deformed vessel head prior to failure. Since the OLHF-4 vessel appears nearly symmetric, this provides some evidence that early deformation in the vessel head is axi-symmetric. Calculation performed by Finnish participants showed that penetration behavior in OLHF-4 could be adequately represented by a 2-D simulation.

On the other hand, location and extent of failure site must be determined by 3-D modeling. The OLHF tests as well as LHF tests demonstrate that rather small asymmetries in temperature or wall thickness could lead to large asymmetries in the overall vessel deformation. Therefore, a 3-D model would be needed to completely characterize the vessel deformation and to predict the location and extent of the failure site. The propagation of the failure and final size of the failure is still an issue; 3-D calculations by CEA, presented in the OLHF 2002 Seminar, show promising results.

The simple SAC models that were evaluated in this program proved quite adequate in predicting vessel failure in the OLHF tests. Various failure criteria, damage functions, and strain rate equations were assessed. For the conditions of these tests, the accuracy of the various models in predicting the test failure times appears to lie within the spread of results obtained by the Finate Element Analysis (FEA) benchmark analysis. This would support the use of such simple models for the purpose of severe accident analysis. However, the correlation for LMP parameters should be based on OLHF property data, as those based on LHF data predict earlier failure.

OLHF Seminar

An OLHF seminar, sponsored by OECD, was held in June 2002 to provide an in-depth review of the project technical capabilities, results, and analyses. Progress in the areas of material characterization, finite element model improvements, modeling vessel penetrations, crack propagation, and simplified models for severe accident analysis, is found in the Proceedings.

Planned Activities

This completes the OLHF experimental activities; the final report will be published as an OECD report.

The data from these tests has been well-qualified and well-archived for future reference. The data is organized into Microsoft Excel spreadsheets with numerous macros for visualization and simple analysis of the data. This project has produced a database that has become an international standard for assessing vessel deformation and failure models.

NUCLEAR ENERGY RESEARCH INITIATIVE

The Secure Transportable Autonomous Light Water Reactor—STAR-LW (IRIS Project)

Primary Investigator: Mario D. Carelli, Westinghouse Electric Company LLC

Collaborators: University of California, Berkeley, USA; Massachusetts Institute of Technology (MIT); Polytechnic Institute of Milan, Italy

Project Number: 99-027

Project Start Date: August 1999

Project End Date: January 2003

Research Objective

This program, currently known as the International Reactor Innovative and Secure (IRIS) project, has the objective of investigating a novel type of water-cooled reactor which can satisfy the Generation IV goals: fuel cycle sustainability, enhanced reliability and safety, and improved economics. The research objectives over the three-year program are as follows:

- First year: Assess various design alternatives and establish main characteristics of a point design.
- Second year: Perform feasibility and engineering assessment of the selected design solutions.
- Third year: Complete reactor design and performance evaluation, including cost assessment.

Research Progress

The original four member, two-country team grew to 11 members in the first year of activity, and further expanded to 18 members from nine countries at the end of the third year, although the French Atomic Energy Commission (CEA [1]) withdrew at the end of the first year and the Japanese JAPC and MHI had withdrawn by the end of the second year. Two more organizations from Brazil, Eletronuclear, and Industrias Nucleares do Brazil, are considering joining the IRIS team. All the added team members work under their own funding and it is estimated that the value of their in-kind contributions in the second year was about $8M, which grew to approximately $12M in the third year. Four universities (University of Tennessee, Ohio State University, University of Michigan and Iowa State University) and two laboratories (Ames and Sandia) also became associated with the program through additional NERI programs and students projects. To date, 72 students have worked or are working on IRIS. By December 2002, 51 IRIS-related graduate theses will have been prepared or are in preparation, and 28 students will have graduated with M.S. or Ph.D. degrees.

The large increase in additional effort, not envisioned in the proposal submitted, has allowed the researchers to significantly exceed their original objectives and to change the outlook for IRIS from a long-term R&D project to a commercially viable design with a deployment target date in the next decade. Several interactions have taken place with NRC at various levels (commissioners, staff, and ACRS) and IRIS pre-application licensing was formally initiated in October 2002.

In response to requests from utilities, IRIS site layout and site bounding information were provided to the ESP (early site permit) program.

The IRIS conceptual design was completed. A summary of program highlights follows.

- The changed emphasis towards a competitive, early deployable reactor has led to two major changes in the IRIS design. First, the reference IRIS size was set at 1,000 MWt (~ 335 MWe), although the same design configuration covers the 100 to 335 MWe range with only modest changes in dimensions. Second, the core design now features a 4.95 percent enriched UO_2 fuel in a 17x17 square array assembly (a 15 x 15 assembly is also possible), very similar to standard Westinghouse pressurized water reactor (PWR) assemblies. Since this fuel has an expected straight burn lifetime of slightly more than four years, with a burn-up of approximately 40,000 MWd/t, it presents <u>no</u> licensing issues. The IRIS core is designed to accept various configurations (8-year straight burn, 8-10 percent fissile UO_2 and MOX fuel;

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4-year straight burn 4.95 percent enriched UO_2; two and three batch 4.95 percent enriched UO_2). The latter are for initial deployment, while the former can be considered for future reloads.

- The IRIS vessel (see Figure 1) includes eight helical steam generators, eight "spool-type" pumps, and pressurizer and internal shields. Six different steam generator designs were evaluated and the Ansaldo helical design was chosen both for its performance and for the fact that it had already been extensively tested in a 20 MWt mockup. The fully internal pumps are based on a design developed for chemical applications. They can be operated in a high-temperature environment and can have large coastdown and run-out capabilities, but have to be qualified for nuclear applications. The pressurizer is of the steam type and the ratio of its volume to reactor power is favorable in being much larger than loop PWRs, thereby allowing very smooth pressure control. The internal shields fill very conveniently into the annular space between the core and the vessel and they reduce the radiation field at the vessel outer surface to the order of 10^{-4} Sv/hr. This has very positive implications for operational and maintenance doses, for long vessel life, as well as for decommissioning and disposal (the "cold" vessel can act as a sarcophagus for the whole reactor internals minus the fuel).
- The concept of "safety by design" (to physically prevent accidents from occurring rather than coping,

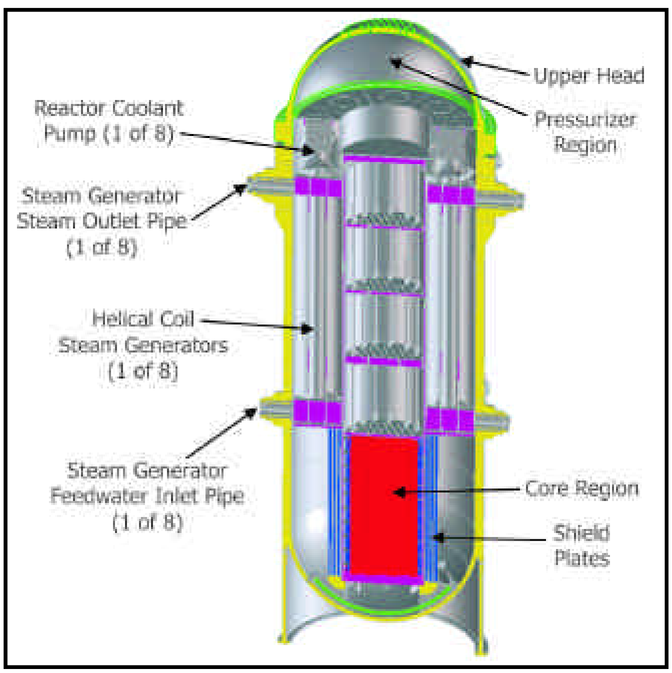


Figure 1. The diagram shows the assembly of the IRIS NSSS.

by active or passive means, with their consequences) has been developed and articulated in detail and its implementation has widely exceeded expectations. Not only are large loss of coolant accidents (LOCAs) eliminated from occurring, as can be expected with all integral designs, but a patented containment design has also practically eliminated small and medium LOCAs as a safety concern. In fact, the core remains fully covered for an extended period of time, several days and possibly weeks, without any safety injection or water make-up. This is made possible by a design that thermo-hydraulically couples the vessel and the containment so that within the first hour after the pipe breaks, the pressures inside the vessel and the outside pressure in the containment equalize, thereby canceling the differential pressure that drives the coolant across the break. Also, the IRIS vessel has no penetrations at and below the core region, and it sits in an open cavity that extends above the core level. Suppression pools are located in the containment and they can double as gravity makeup, even though they are not needed. Decay heat is removed by four diverse (three independent) systems: eight steam generators, four natural circulation heat exchangers located outside the containment, and surface (air and water) containment-cooling.

Loss of flow accidents (LOFAs) have no significant consequences because of the pumps' characteristics and redundancy, as well as the substantial degree of natural circulation. Steam generator tube rupture accidents have lower probability and more benign consequences since the tubes are in compression (with primary coolant outside) and are designed for zero internal pressure.

The conclusion is that IRIS is a water reactor design where primary coolant-related accidents are of no major concern, and of the eight Class IV accidents typically considered in the advanced passive designs safety analysis reports (SARs), only one (refueling accident) remains as a Class IV accident and even that, with a much lower probability. All the others either no longer apply, or can be reclassified at Class III or lower.

- An IRIS model using the RELAP5 code was completed to perform initial plant safety assessment and was verified in a preliminary steady state and transient qualification. All the relevant transient events and accidents typically reported in a Safety

Analysis Report were assessed, including (but not limited to) steam system piping failure, feed system piping failure, loss of offsite power, turbine trip, loss of flow, locked rotor, reactivity anomalies, steam generator tube failure, small break LOCA, and anticipated transients without scram.

In accordance with standard procedures, the system code (RELAP) is coupled with subchannel and neutronic analysis codes when required by the specific event considered. CFD analyses of selected portions of the pressure vessel have been performed to verify mixing phenomena in the IRIS system. For the analyses of small break LOCA, the strong coupling between vessel and containment during most of the event duration has required development of new approaches for the system analyses. While different solutions have been explored, a thermal-hydraulic coupling of RELAP (for reactor coolant system analysis) and GOTHIC (for containment analyses) was identified as the most promising approach and used in the analyses.

Models and results of the analyses have been collected in a preliminary plant safety assessment document to be submitted to the NRC as part of the IRIS pre-application review. Probabilistic Safety Assessment (PSA) analyses have been initiated, with a preliminary assessment of events and fault trees.

- Substantial work has been completed to support the IRIS goal of a 48-month interval between maintenance shutdowns. This, coupled with the core lifetime of about four years without refueling, will yield very high capacity factors and significantly reduce the operating and maintenance (O&M) costs. A previous effort was performed by MIT to investigate the feasibility of extending the maintenance interval in a commercial PWR from 18 to 48 months. A total of 3,743 maintenance items were identified for the 18-month cycle, 1,206 to be performed on-line and 2,537 off-line during the scheduled outage. The MIT study showed that most of the 2,537 off-line items could be deferred to 48 months or be performed on-line. Only 54 items in various categories (e.g., relief valves, motor operated valves, pumps) remained outstanding, as they still required an 18-month interval.

  Building on this study, the unresolved items were examined for their applicability to IRIS (e.g., pump oil lubrication obviously does not apply to the reactor coolant lubricated internal spool pumps). Only seven items in five categories were finally identified as still outstanding impediments to a 48-month maintenance interval in IRIS. None of these are considered showstoppers and work is in progress for their resolution. An additional category was identified for items that could be tested on-line, but would require a reduced power level for the test.

- As part of the ESP input, two potential arrangements of multiple IRIS modules were identified: one (1,000 MWe total) consisting of three modules "in a string" with staggered construction start and one (1,350 MWe total) consisting of two twin units where each unit has two modules sharing almost all auxiliary systems.

- A market analysis and preliminary top-down cost estimate was performed, confirming the competitive attractiveness of IRIS, both in developed and emerging countries. The total cost of electricity was on the order of $0.03/kWh, with a best estimate of $0.0285/kWh.

- The necessary testing program to confirm the operational and safety characteristics of IRIS is being outlined. A preliminary assessment is underway, which includes preparation of PIRTs (Phenomena Identification and Ranking Table), identification of required tests, specification of parameters, assessment of similitude analyses, preparation of test plans, and identification of test facilities.

- Risk informed regulation is being assessed as an option for the IRIS licensing, with the objective of demonstrating that IRIS can achieve the stated Generation IV goal of eliminating the need for offsite emergency response planning.

Planned Activities

IRIS development does not end with the conclusion of the NERI three-year program. The IRIS consortium is proceeding with detailed design and analyses, the NRC licensing process has been initiated, and the ESP program will continue with IRIS participation. In reality, IRIS development will proceed on a higher scale to fulfill the objective of producing a successful commercial entry in the next decade.

NUCLEAR ENERGY RESEARCH INITIATIVE

Monitoring and Control Technologies for the Secure Transportable Autonomous Reactor (STAR)

Primary Investigator: Hussein S. Khalil, Argonne National Laboratory

Collaborators: Lawrence Livermore National Laboratory (LLNL); Texas A&M University

Project Number: 99-043

Project Start Date: August 1999

End Date: September 2002

Research Objectives

A new reactor and fuel system concept designated as the Secure Transportable Autonomous Reactor (STAR) has been proposed for meeting the needs of developing countries for small, economical nuclear power stations while at the same time addressing proliferation concerns. This NERI project supports this goal through development of operations-monitoring, and control and remote surveillance strategies that exploit the passive safety and autonomous operation attributes of the STAR plant. It also entails development and demonstration of advanced technologies for implementing these strategies to assure operational reliability and security of nuclear materials.

Specific objectives of the research are to simplify active control and safety protection systems; minimize reliance on on-site operating staff; and assure high levels of operational safety, reliability, and facility security. Research tasks include evaluating the ability of candidate STAR plants to operate autonomously with minimal reliance on active control for load adjustment and burnup reactivity compensation, identifying design and operating features that enhance operational autonomy and passive safety, developing simplified control strategies on the basis of the passive plant response, and developing and demonstrating computer-based technologies for remote monitoring of operational and safeguards information at centralized surveillance facilities.

Although proposed in the framework of the development effort for the STAR system, the research addresses issues of fundamental importance to the operation of passively safe and autonomous plants. Resolution of these issues will increase the immunity of passively safe plants to operator and control system errors and provide a technical basis for reducing the cost of plant control and safety protection systems.

Research Progress

Accomplishments in the first two years of the project include the following:

Autonomous Operability of STAR Designs: Design options were identified for achieving autonomy of operation in the lead-bismuth eutectic (LBE) cooled STAR-LBE concept (see Figure 1). The approach for increasing autonomy focuses on the use of inherent properties of mechanical, hydraulic, thermal, and neutronics reactor systems, which are determined by the choice and arrangement of reactor materials. One consequence of the enhanced use of intrinsic feedbacks is reduced need for plant actuators.

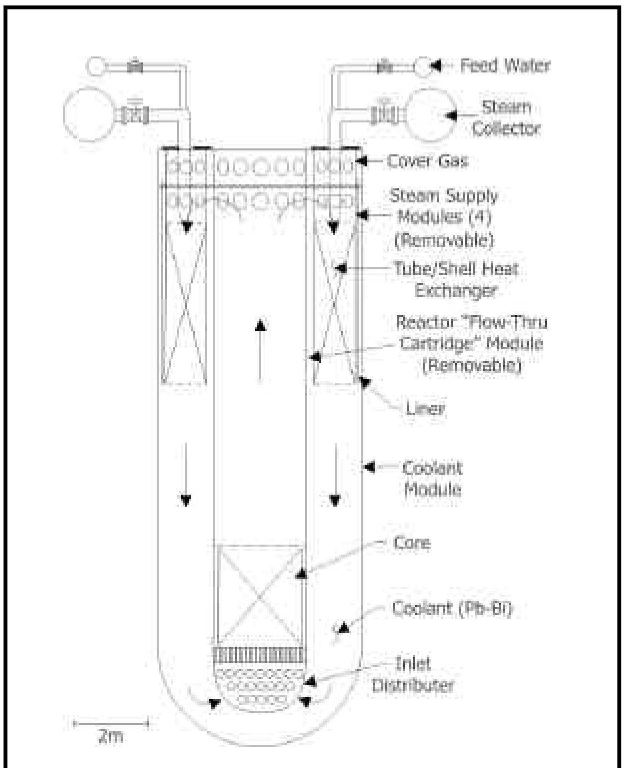


Figure 1. The diagram shows a STAR-LM simplified, modular, small reactor featuring a flow-through fuel cartridge.

Buoyancy-driven natural circulation eliminates primary pumps, temperature reactivity feedbacks replace rods for changing power, and internal conversion eliminates rods otherwise needed for compensation of reactivity excess at beginning of cycle. Substantial flexibility exists in the form of fuel type, core layout, reactor steel selection, primary loop layout, steam generator design, and selection of water-side equipment. It was found that inherent feedbacks can be used to make the STAR-LBE plant virtually self-regulating with respect to load change, reactivity control with burnup, and unprotected upsets. Startup, however, was found to require the insertion of rod worth to overcome reactivity associated with a negative power coefficient. The negative coefficient is dictated by stability requirements and is a technical specification. Other than needing a control system to deliver this worth, inherent feedbacks can provide for operation that is virtually free of the need for active control elements. Moreover, the analsis of reactivity effects for STAR-LBE with metallic fuel showed that these are no combinations of operator or control system errors that could cause reactor temperatures to exceed safe limits.

The possibility of using reactivity feedbacks to maneuver power and avoid the use of control rods was also explored for a natural-circulation boiling water version of the STAR concept (STAR-BWR). The first step in designing a STAR-BWR that does not use control rods to accomplish load following is to demonstrate that the desired range of steady-state powers is achievable. This has been the focus of the current effort. The base case 100-MWe plant used conventional uranium oxide fuel and was found to be capable of reaching a minimum power of only 82 MWe during load following. The alternative 100-MWe plant was capable of reaching 37.5 MWe. The alternative, however, assumes a very high fuel conductivity, indicative of metallic fuel, and a doubling of the ratio of the void coefficient to the temperature coefficient. Additional study of the thermal and neutronics aspects of a BWR and its fuels would thus be necessary to determine what is achievable in practice. Moveover, dynamic behavior and stability implications of reliance on passive control and natural circulation cooling should be examined in future work.

<u>Remote Monitoring System Design:</u> A design for a remote monitoring system was developed to meet anticipated security and operational monitoring requirements for a STAR plant. In this system, key plant security signals are acquired, digitized, encrypted, and sent via the communication system to the remote monitoring site. Security requirements were defined assuming all persons with normal access to the plant may be a threat. Accordingly, the security requirements start at the sensors and extend through the communications system. Many of the sensors have dual use for security and operation (e.g., reactor temperatures and flux levels), while others are dedicated to security (e.g., motion or volume sensors, video cameras, and so forth). This data is transmitted to the remote site with a sampling rate appropriate to the data, but no more frequently than once each second. This data would be monitored at the remote site essentially in real time and would be stored for trending displays and additional analyses. Features to thwart potential attempts to subvert the monitoring system include limiting access to system components and use of a robust combination of tamper detecting devices and information analysis capabilities based on "machine intelligence" techniques.

Key features of the monitoring system design have been implemented in cooperation with Texas A&M University. A satellite-based network was established to allow remote observation of the Texas A&M Nuclear Science Center Reactor, and to provide integrated voice video and data transmission between LLNL and the reactor.

<u>Demonstration of Remote Reactor Monitoring:</u> A capability was developed and implemented for remote monitoring of the Texas A&M University Nuclear Science Center Reactor (NSCR). This capability uses standard hardware components and a software package (LabVIEW) available from National Instruments Co. for data and image acquisition, information analysis and storage, and remote monitoring of this information on the Internet using either browser or LabVIEW software. The new hardware and software were installed to provide a capability for Internet- or satellite-based viewing of the NSCR main control room console, the fuel and water system temperatures, the reactor log and linear power readings, control rod positions, and Facility Air Monitoring System data. In addition, a video-based surveillance system has been designed and implemented for remote viewing of the NSCR core. This surveillance system is illustrated in Figure 2. The required video equipment was procured and interfaced with the LabVIEW software for display at the monitoring site.

In addition to simplifying the control and safety protection systems, the STAR plant design and the proposed approach to plant control promise to allow uninhibited deployment of computer-based operations

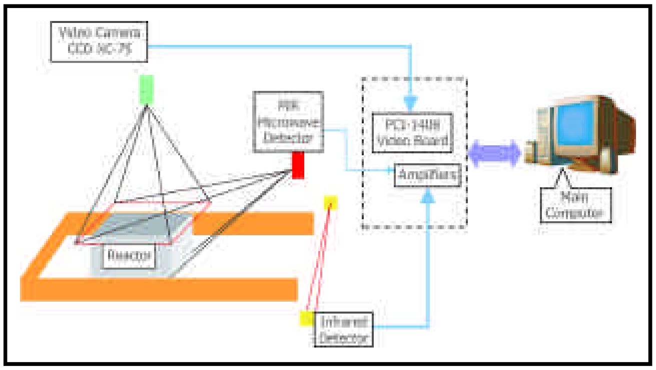


Figure 2. The diagram shows the video monitoring system implemented for the NSCR at Texas A&M University.

support systems that exploit modern digital-based computing and information management technologies. Key roles (functionality) identified for these technologies are to (a) generate, substantiate, and graphically display sensor information required to describe the state of plant systems, and (b) exploit this information to maximize plant availability, optimize maintenance activities including those requiring the dispatch of specialists to the plant site, and assist on-site operators to cope safely and effectively with potential upsets. Software reliability is the key challenge identified until now for the deployment of these digital technologies.

To address this challenge and provide the needed functionality, operator support systems should satisfy key requirements including inherent portability, ability to accomodate unforeseen events and amenability to generic verification and validation. These requirements could be satisfied by employing a first-principles modeling approach augmented by the use of measured data. This approach would overcome the limitations of hard-wired course/symptom methods that are inherently plant- and process-specific.

Planned Activities

The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

Risk Informed Assessment of Regulatory and Design Requirements for Future Nuclear Power Plants

Primary Investigator: Stanley E. Ritterbusch, ABB-Combustion Engineering

Collaborators: Sandia National Laboratories; Idaho National Engineering & Environmental Laboratory; Massachusetts Institute of Technology; North Carolina State University; Duke Engineering Services; Eagan & Associates

Project Number: 99-058

Project Start Date: August 1999

Project End Date: December 2002

Research Objective

Research under this project addresses the barriers to long-term use of nuclear-generated electricity in the United States. Studies being performed by the Electric Power Research Institute on the cost of coal, gas, and nuclear-generated electricity have identified that to be competitive, the cost for the nuclear option would have to decrease to the range of $0.03/kilowatt-hour over the next two decades. Correspondingly, the total plant capital cost of a typical Advanced Light Water Reactor (ALWR) would have to decrease by about 35 percent to 40 percent and the construction schedule would have to be shortened to about three years.

Research Progress

Shortly after initiating this project, team members established the principal strategies required to achieve the project's cost-reduction goals. It was agreed that a very basic and significant change to the current method of design and regulation was needed. That is, it was believed that the cost-reduction goal could not be met by fixing the current system (i.e., an evolutionary approach) and a new, more advanced approach for this project would be needed. It is believed that a completely new design and regulatory process would have to be developed-a "clean sheet of paper" approach. This new approach would start with risk-based methods, would establish probabilistic design criteria, and would implement defense-in-depth only when necessary to (1) meet public policy issues (e.g., use of a containment building no matter how low the probability of a large release) and (2) address uncertainties in probabilistic methods and equipment performance. This new approach is significantly different from the Nuclear Regulatory Commission's (NRC's) current risk-informed program for operating plants. For the new approach, risk-based methods are the primary means for assuring plant safety, whereas in the NRC's current approach, defense-in-depth remains the primary means of assuring safety.

The primary accomplishments in the first year (Phase 1) of this project include

(1) The establishment of a new, highly risk-informed design and regulatory framework;

(2) The establishment of the preliminary version of the new, highly risk-informed design process;

(3) Core damage frequency predictions showing that, based on new, lower pipe-rupture probabilities, the design of the emergency core cooling system equipment can be simplified without reducing plant safety; and

(4) The initial development of methods for including uncertainties in a new integrated structures-systems design model.

Under the new regulatory framework, options for the use of "design basis accidents" were evaluated. Whereas, in the current regulatory process, the design basis accidents and their evaluation are primarily deterministic, it is expected that design basis accidents would be an inherent part of the Probabilistic Safety Assessment for the plant and their evaluation would be probabilistic.

Other first year accomplishments include

(1) The conversion of an NRC database for cross-referencing NRC criteria and industry codes and standards to Microsoft 2000 software;

(2) An assessment of the NRC's hearing process,

which concluded that the normal cross-examination during public hearings is not actually required by the U.S. Administrative Procedures Act;

(3) The identification and listing of reliability data sources; and

(4) Time spent interfacing with other industry groups [e.g., Nuclear Energy Institute (NEI) and International Atomic Energy Agency (IAEA)] and NRC at workshops for risk-informing regulations.

The foregoing tasks were continued during the second year (Phase 2) of this project. In addition, Westinghouse signed an agreement with the Korea Power Engineering Company (KOPEC) for their participation in this project. Major benefits of this cooperation include the Korean design and operating experience that is brought to the project, and the completion of a relevant work scope by KOPEC personnel. This agreement was reviewed and approved by other project participants as well as the Department of Energy.

The major accomplishments during the second year consisted of

(1) Issuance of the final report for Subtask 1.1, "Identify Current Applicable Regulatory Requirements [and Industry Standards]";

(2) Issuance of the final report for Subtask 1.2, "Identify Structures, Systems, and Components and Their Associated Costs for a Typical Plant";

(3) Extension of the new, highly risk-informed design and regulatory framework to non-light-water-reactor technology;

(4) Completion of more detailed thermal-hydraulic and probabilistic analyses of advanced conceptual reactor system/component designs;

(5) Initial evaluation and recommendations for improvement of the NRC design review process; and

(6) Initial development of the software format, procedures, and statistical routines needed to store, analyze, and retrieve the available reliability data.

In Phase 2 of the project, several reports and documents were produced as planned. Final reports for Subtasks 1.1 (regulatory and design criteria) and 1.2 (costs for structures, systems, and components) were prepared and issued. A final report for Subtask 1.3 (Regulatory Framework) was drafted with the aim of issuing it in Phase 3 (Year 3). One technical report was produced for Subtask 1.4 (methods development) and two technical reports were produced for Subtask 1.6 (sample problem analysis). An interim report on the NRC design review process (Subtask 1.7) was prepared and issued. Finally, a report on Subtask 2.2 (database weaknesses) addressed the initial development of a new database to track reliability data.

During the third and final year (Phase 3) of this project, work was completed on Subtasks 1.3 (regulatory framework), 1.6 (sample problem analysis), Subtask 1.7 (regulatory analysis), Subtask 1.8 (industry and NRC coordination), and Subtask 2.3 (reliability data improvements). Also during the third year, more detailed thermal-hydraulic and probabilistic analyses of the advanced conceptual designs were completed. The results of these efforts will be documented in subtask reports and in the final report for this project. The regulatory framework was updated to reflect the extension of new risk-informed methods to non-LWRs (Subtask 1.3), and supporting analyses for a pebble-bed reactor design were completed (Subtask 1.6). Subtask 1.6 also covered additional thermal-hydraulic analysis for the current generation LWR loss of coolant accident. Work on software to facilitate access to reliability data was completed in Subtask 2.3.

Planned Activities

Final subtask reports for the subtasks completed in Phase 3 (indicated above) will be issued. Also, the Phase 3 Annual Report and the project's Final Report covering all three years will be issued.

NUCLEAR ENERGY RESEARCH INITIATIVE

Demand-Driven Nuclear Energized Module

Primary Investigator: Gary T. Mays, Oak Ridge National Laboratory (ORNL)

Project Number: 99-064
Project Start Date: August 1999
Project End Date: September 2002

Research Objectives

The Nuclear Energizer Module (NEM) is a reactor concept designed to take advantage of newly developed graphite foam, which has enhanced heat transfer characteristics and excellent high-temperature mechanical properties and will provide an inherently safe, self-regulated source of heat for power and other potential applications.

The principal objectives of the project are to (1) develop a reactor concept with a target power level of 500 kW(th) that is naturally load following, inherently safe, optimized via neutronic studies to achieve near-zero reactivity change with burn-up, and proliferation-resistant; (2) prepare and appropriately characterize the physical properties of the graphite foam; (3) conduct irradiation studies of the graphite foam to determine any effects on structure, dimensional stability, thermal conductivity, and thermal expansion; (4) simulate the overall performance of the reactor concept in terms of operations, safety, stability, and thermal characteristics; and (5) develop a physical model using the graphite foam and electric heaters to benchmark computer models. The general application targeted for this concept is a design that is easily deployable to supply power to remote and/or harsh environments.

Research Progress

This project has four major tasks, in the following areas:

- Neutronics
- Materials testing and evaluation
- Simulation and thermal analysis
- Power conversion

A brief summary of progress in each of these areas follows.

Neutronics Analyses: The reactor core consists of a right circular cylinder of graphite foam impregnated with uranium carbide. The radius of the cylinder is 75 cm. The cylinder is divided into two zones: a central unpoisoned region having a radius of approximately 13 cm and an outer annular region in which the foam is impregnated with cadmium at a Cd density of 86 milligrams per cm^3. The $C/^{235}U$ ratio is 39. The uranium [20 percent enriched U; critical mass of about 4 Metric Tons of Uranium (MTU)] exists as UC_2. During normal operation, the cadmium is relatively innocuous (the reactor has a fast spectrum). However, should there be ingress of water to the core, the cadmium acts as an effective neutron poison causing the reactor to be subcritical. The temperature coefficient of the fueled-foam is expected to be between -0.2 and -0.3 pcm/°K.

The core is surrounded by a graphite reflector (i.e., normal graphite, not graphite foam). Uranium-impregnated foam would be encased in a "super-alloy" steel. The reflector/clad interface temperature is expected to be limited to 900°K. The reflector thickness (considering reactor weight, size, and cost of fuel) is approximately 30 cm. Shutdown/startup of the reactor would be achieved by control elements in the radial reflector, outside the core clad.

Materials Testing: Four capsules, each with three foam samples plus a SiC temperature monitor, were irradiated in the High Flux Isotope Reactor at ORNL. Two capsules were used for the in-plane samples, and the other two were used for the out-of-plane samples. One in-plane and one out-of-plane capsule received an irradiation dose of 2.6 displacements per atom (dpa); while the other in-plane and out-of-plane capsule received an irradiation dose of 0.3 dpa. The location of the capsules within the hydraulic tube was such that the variation of the neutron flux was less than 15 percent from capsule to capsule.

Evaluation of the SiC temperature monitors allows the determination that the actual irradiation temperature of the capsules was approximately 820°C; this temperature is considerably higher than the planned irradiation temperature of 600°C. Observation of irradiated samples under a Scanning Electron Microscope (SEM) showed no apparent changes or damage, i.e., samples conserved their structural integrity during irradiation.

Thermal conductivity measurements and annealing studies of irradiated samples showed that thermal conductivity decreased as the irradiation dose increased. This effect is consistent with typical results for graphite samples. Annealing of irradiated foam samples to 1,000°C and 1,200°C lead to recovery of their thermal conductivity. Figure 1 shows the results of the thermal conductivity measurements as a function of the measurement temperature, for sample OP-9, irradiated at 2.6 dpa.

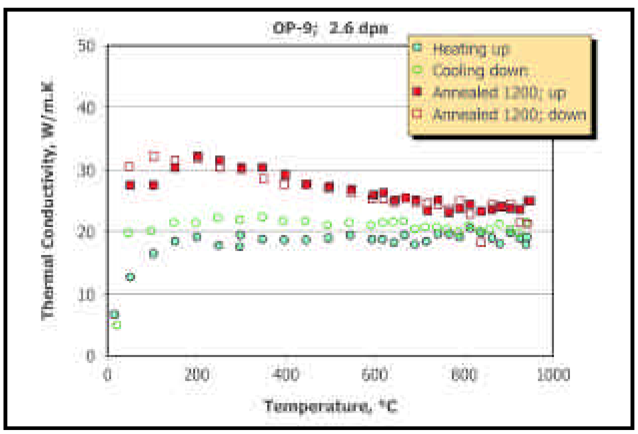


Figure 1. The graph represents thermal conductivity measurements of sample OP-9 after irradiation and after annealing to 1,200°C.

Simulation and Thermal Analysis: A variety of computations have been performed to assess the temperature distributions for candidate reactor configurations. Computations indicate that for the target 500 kW(th) fission heat generation, cooling on the top and bottom circular boundaries should be considered in addition to the cylindrical surfaces presently chosen. Heat removal at the core/reflector boundary instead of at the reflector's external boundary may also be required.

Two-dimensional computations of temperature distributions inside the core and reflector for gas cooling are being performed for the present configuration of heat removal only at the reflector's external cylindrical surface, heat removal only at the core/reflector's internal cylindrical surface, and heat removal through the bottom and top circular surfaces in addition to the cylindrical surfaces.

Power Conversion: The reference thermal power conversion system consists of the heat transfer geometry surrounding the nuclear core cladding and the closed Brayton cycle components that convert the thermal power generated by the reactor into electrical power. Heat transfer fins are necessary to increase the effective heat transfer area of the core clad surrounding the nuclear-fueled region. The fins will be made of the same material, a super-alloy steel as the core clad, and the clad surrounding the reflector will serve as an outer shroud to contain the cooling flow. Based on optimization studies, the fin geometry is as follows: fin height of 10 mm, fin width of 5 mm, and fin spacing of 10 mm. Using this cooling geometry, the working fluid pressure drop in the finned area is less than 1 percent.

The working fluid of the recuperated closed Brayton cycle system chosen is a mixture of helium and xenon, with a molecular weight of 40 (HeXe40). If pure helium were chosen as the working fluid, the relatively high thermal conductivity would result in the recuperator having a minimal necessary heat transfer area as compared to other gases, thereby minimizing the overall volume envelope of the system. However, pure helium also has a high specific heat that would limit the temperature change and therefore the pressure ratio across a stage. This stage pressure ratio limit would increase the number of turbine and compressor stages and result in complicated turbomachinery. Thus, a compromise is necessary between the small recuperator and complicated turbomachinery for pure helium and the larger recuperator with simple turbomachinery of HeXe40. Closed Brayton cycle turbomachines have been successfully designed and tested for space applications using HeXe40 in the electrical power range of interest (~100 kWe). Therefore, the HeXe40 turbomachinery design was chosen. Because the working fluid is clean and inert, relatively long full power operating intervals of a decade can be expected.

Because of core/clad interface temperature limitations, the turbine inlet temperature has been specified as 900°K. Since a terrestrial location is postulated for this reactor concept, a compressor inlet temperature of 310°K is specified. The ultimate heat sink of water, forced convection air, and natural convection air were analyzed. While water would result in the smallest size heat-rejection heat exchanger and perhaps a lower compressor inlet temperature, the presence of water vapor near the active core during an upset transient is a nuclear criticality consideration. A natural convection heat sink design would require an air chimney and have a large heat rejection heat exchanger. Thus, for simplicity, the forced convection air design is chosen as the ultimate heat sink.

Recuperator effectiveness is a measure of the actual heat transferred to the maximum possible from the hot side to the cold side of the recuperator. A reasonably achievable effectiveness is 95 percent which is within the existing industrial capabilities for manufacturing.

The system pressure drop is defined as the sum of the individual component fractional pressure drops. The component fractional pressure drop is defined as the actual pressure loss divided by the absolute working pressure of the component. This system fractional pressure drop determines how much of the turbine output is absorbed as parasitic pressure loss. A reasonable and realistic value of 5 percent is chosen, where 1 percent is budgeted for the core heat transfer fins and 4 percent for the remainder of the closed Brayton cycle. In order to keep the system as uncomplicated as possible, the compressor will not be intercooled.

Industrial firms under contract to NASA have designed HeXe40 turbomachinery appropriate for the electrical power range of interest here that use a radial flow design, which has been selected for this application. The compressor discharge pressure is 1.48 MPa (14.5 atm or 200 psig). This pressure, the highest in the system, is relatively low and results in a simple pressure containment boundary. A reasonable compressor polytropic efficiency of 85 percent and a turbine polytropic efficiency of 90 percent are achievable for this size turbomachinery.

Because the NASA turbomachinery was initially designed to use an 80 percent effective recuperator, the pressure ratio across the compressor is 2.2. Using this pressure ratio of 2.2 with the above cycle parameters results in a cycle thermal efficiency of 27 percent with a HeXe40 mass flow of 4.33 kg/s. Thus, for the proposed core thermal power of 500 kW, 135 kW of shaft power is available using the existing NASA turbomachine design. With a generator efficiency of 95 percent, the electrical power resulting from using the existing turbomachinery design is 128 kW.

However, since a 95 percent effective recuperator is sized here, the pressure ratio across the compressor that results in the optimum cycle efficiency is 1.65. Using this pressure ratio with the above cycle parameters results in a cycle thermal efficiency of 30 percent with a HeXe40 mass flow of 6.28 kg/s. Thus, for the proposed core thermal power of 500 kW, 150 kW of shaft power is available. With a generator efficiency of 95 percent, the electrical power resulting from this design is 143 kW. The existing NASA turbomachine design would have to be modified to the lower pressure ratio, which should be easily accommodated.

Planned Activities

The remaining activity involves the preparation of a final report documenting the research and analysis performed and describing the preconceptual design for this reactor system.

NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Advanced Technologies to Reduce Design, Fabrication and Construction for Future Nuclear Power Plants

Primary Investigator: Camillo A. DiNunzio, Framatome ANP DE&S

Collaborators: North Carolina State University; Massachusetts Institute of Technology; Sandia National Laboratories; Westinghouse Electric Company Nuclear Systems; Korean Power Engineering Company, Inc. (KOPEC)

Project Number: 99-077

Project Start Date: August 1999

Project End Date: November 2002

Research Objectives

The goal of the Design, Procurement, Construction, Installation, and Testing (DPCIT) project team is to identify methods that can deliver a 40 percent reduction in capital cost and scheduling time for a future nuclear power plant. Given the starting point of the Advanced Light Water Reactor (ALWR) program's benchmark of $1,500 per kW installed, and a 60-month construction schedule, the targeted reductions translate into a power plant built for $900 per kW and 36 months from first concrete to fuel load. Ideally, these techniques and innovations could be demonstrated in near-term reactor plants and then incorporated into the planned Generation IV reactor plants, which are under conceptual development. Most of the innovations planned in this project should be applicable to most reactor technologies since the concepts under consideration are not linked to a specific reactor design. However, the proposed reductions will be demonstrated using pressurized water reactor technology.

Research Progress

The DPCIT project represents a merger of information technologies and supply chain management principles with design and construction improvements. The project focused on processes, and along the way adopted the appropriate tools to execute the process. Figure 1 provides a visual image of the intent of the DPCIT project.

The project consisted of tasks performed over a three year period, 1999-2002.

Year 1 Summary

During the first year of project activities (1999-2000), the project team identified initial strategies for reducing capital costs and scheduling. A key insight gained by the team is that back-fitting new technologies into pre-existing designs does not offer as significant a payoff as working with new designs as they are being created. This favors the Generation IV advanced nuclear plants since these plants are in the process of incorporating the kinds of improvements this project has examined, or there is the potential to incorporate these insights before their designs are finalized.

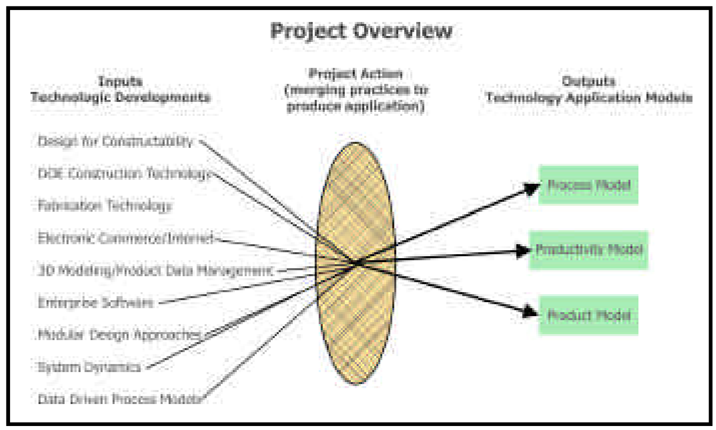


Figure 1. DPCIT project overview

Year 2 Summary

During the second year's activities, initiatives pursued included examining the potential impact of information technology on physical construction activities (the impacts of information technology on the design side of construction were examined in year 1); evaluating the potential for project management concepts to serve as advanced tools to help assure on-budget and on-schedule performance of a complex project such as a nuclear plant; and examining Cost Risk Modeling as a mechanism for assessing the viability of potential cost and schedule reduction techniques. Additionally, mechanisms were explored to remove excess conservatism and reduce error while shrinking the time between design and analysis activities, and methods were evaluated for removing excess seismic margins and simplifying containment construction.

Year 3 Summary

In year three, the final phase of this project, work focused on the following areas:

Containment and Structural Simplifications: Work continued on evaluating excess margins and simplifying structural systems. While reduced loads are likely to result by developing simpler alternative structural systems, a trade-off exists between the reduction of seismic margins and risk. Any decision regarding structural simplification should be focused on addressing this trade-off. During year 3 of this project, a decision support system (DSS) was developed for evaluating this trade-off. A case study was developed for simulation-based design, incorporating formal optimization tools and their limitations in an automated process.

A case study for application of the DSS to the design of snubbers and their locations in a piping system was conducted. The genetic algorithms are being used to not only arrive at the most optimal solution but also evaluate the design trade-off. The Modeling to Generate Alternatives (MGA) technique would then be used to evaluate alternative designs that are near optimal but different.

More significant progress than was expected was made in the development of a decision support system. In addition, its application was illustrated by a case study. This illustration was undertaken for two reasons: (1) In the opinion of the researchers, a DSS forms a key element of the process envisioned for DPCIT project; (2) the work had to be reorganized since there was insufficient information for extending the work in Year 2 on categorizing and collecting piping cost data. (It was not possible to develop similar cost data for piping and concrete structures in advanced plant designs.)

Solid Modeling-"Design to Analysis" Tool: The collaborators successfully completed a preliminary version of a "design-to-analysis" tool that converts solid models of a pressurizer and its piping systems in a computer-aided drawing (CAD) package into a finite element mesh ready for analysis. This finite element model, which incorporates the dynamic coupling between the pressurizer and its piping systems, can be used to optimize the design of piping supports as a significant procedure to reduce cost.

Seismic analyses of the finite element model of the pressurizer assembly will be performed with the appropriate floor time histories of seismic accelerations provided by Framatome ANP DE&S. The piping design details of anchor locations and constraint conditions will also be provided by Framatome ANP DE&S and incorporated into the finite element model (see figure 2).

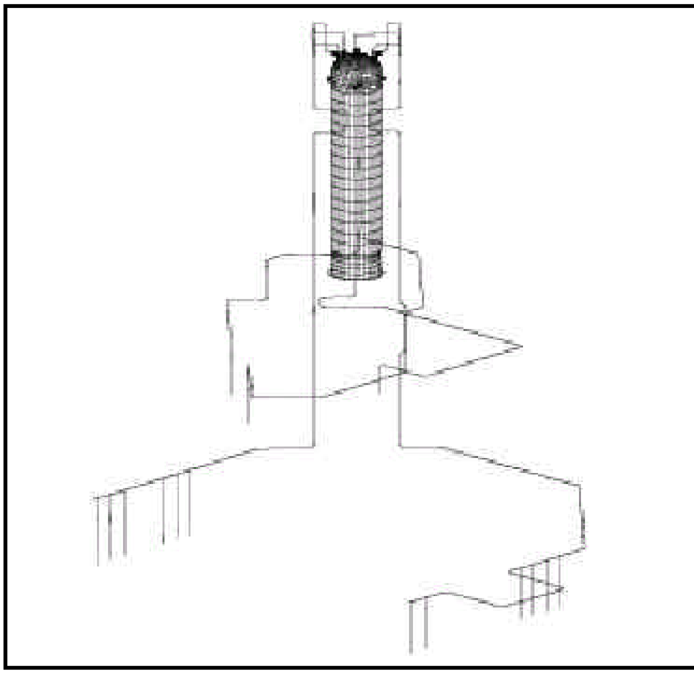


Figure 2. Coupled Model of Pressurizer Assembly and Piping Systems

4D Visualization Modeling: The team identified strategies used to reduce costs associated with the Design, Procurement, Construction, Installation and/or Testing phases of the Korean Standard Nuclear Plant (KSNP) Project, using "4D visualization." Construction data from the KSNP project was collected and the schedule reduction method used was reviewed. A KSNP 4D visualization model was developed and the design model was linked with the construction schedule for the facility's entire power block (i.e., Containment, Auxiliary, and Turbine Building).

Additionally, 4D simulation movie clips were prepared for each building and for the whole power block (see figure 3).

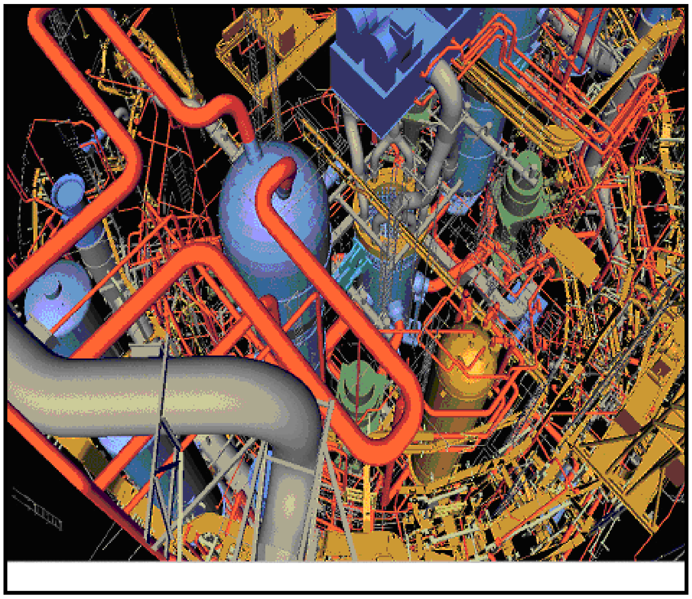


Figure 3. Inside of KSNP Containment Building

Capital Cost Model Development and Testing: The objective of this effort, conducted during the first and second year of the project, was to develop a capital costing model that addressed the uncertainty in the potential savings in both costs and project duration, separate from other research activities within this project. A tool was sought for ranking various options in terms of their ability to reduce capital costs, and for developing a methodology for converting uncertainty to risk. In FY 2001, the methodology was tested using a simplistic example. The process used a work breakdown structure (WBS) to build a factored estimate and schedule. Impacts and range of the factors and schedule durations were assessed using multiple attribute utility analysis. These were all combined in a Monte Carlo analysis to generate a probabilistic distribution of final costs and durations. Ranking of options is achieved by assessing the impact on cost and schedule by individually removing a proposed savings scheme from the model.

The final report included the development of a factored costing model and expansion of the project understanding of the impact of external constraints on project duration and final cost.

Project Management Cost and Schedule Modeling: Initial versions of two advisory systems for project management have been successfully developed: 1) A System Dynamics model that uses the deterministic approach and 2) a Bayesian Belief Network (BBN) model that uses the probabilistic approach. These two systems have been benchmarked with real-case information and showed good agreement between system predictions and real case data. They also incorporate one of the very important issues in the real project management area that has not been included explicitly in the decision-making process: the long-term benefit (or penalty) of management actions to a project. Additional results derived by the developed advisory systems were benchmarked. These results were compared to real case data or to the opinions of experts, to obtain the real case data. In addition, the "debugging" of the developed interface programs was continued, and more tests performed with the advisory systems (see figure 4).

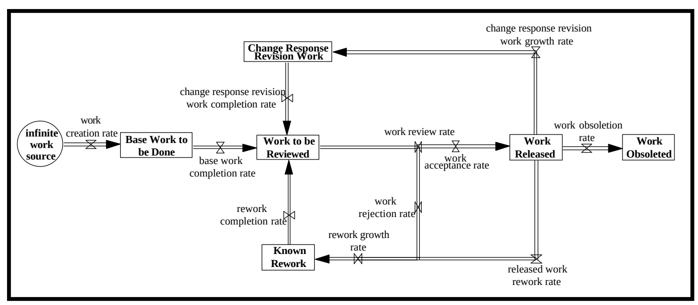


Figure 4. Work flow diagram within a particular stage of the project

Planned Activities

The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

Innovative Chemithermal Techniques for Verifying Hydrocarbon Integrity in Nuclear Safety Materials

Primary Investigator: L. Mason, Pacific-Sierra Research Corporation

Collaborators: University of Virginia; University of Maryland

Project Number: 99-094

Project Start Date: August 1999

Project End Date: August 2002

Research Objectives

This research and development program is designed to explore new methods of assessing the current condition and predicting the remaining life of critical hydrocarbon materials in nuclear power plant environments. Of these materials, Class "1E" safety cable insulation is the primary focus. Additionally, materials for o-rings, seals, and lubricant products designed for nuclear applications will also be studied. This three-phase applied research program is providing industry with new, innovative methods and reference data to conduct pragmatic programs to monitor the condition of materials for a wide range of polymer-based products. Key research objectives include development of a material-condition-monitoring database, optimization and standardization of various testing procedures, implementation of proven engineering development methodologies, and analyses of inter-technology correlations.

Milestones are defined by a five-task, work-breakdown schedule. Task 1 encompasses conducting front-end R&D program activities and planning additional tasks. Task 2 concerns identifying and reporting on subject cable materials, and their acquisition, accelerated aging, and testing by a suite of chemithermal methods. Tasks 3 and 4 comprise similar objectives for, respectively, o-rings, seals, and lubricant materials. Task 5 concerns required programmatic documentation. Under Phase 1 of this project, research progress to date fell mainly within Tasks 1, 2, and 5.

Research Progress

The greatest fraction of Phase 1 resources for this project was expended during conduct of Task 1, the "Research and Development Plan." The next largest fraction of resources was applied to achieve progress on the first research task (Task 2), "Polymer Cable Insulation," specifically described below. In Task 1, personnel requirements were determined and fulfilled; and a new laboratory space was commissioned. Spare parts and routine supplies were acquired for three chemithermal analyzers: a Perkin-Elmer TGA7 (thermogravimetric analyzer); and two Perkin-Elmer DSC7s (differential-scanning calorimeter) instruments, one with an available high-pressure (up to 600 psig) cell accessory. Computer hardware and software upgrades included Perkin-Elmer software upgrades and an operating platform upgrade to Windows NT. These intense activities significantly expanded the laboratory's capability to simultaneously operate the suite of analyzers available at that time, as well as future additions, which now include a Perkin-Elmer TMA7 (thermomechanical analyzer) and a Perkin-Elmer Fourier transform infrared (FTIR) analyzer, the latter being configured to routinely analyze thermogravimetric analysis (TGA) combustion gases. Development of three research task plans proceeded as expected, and were focused, respectively, upon the subject materials: 1E electric cable insulation materials, o-ring and seal materials, and lubricants. Activities for each task included the following: identification of critical materials, acquisition, sample preparation, accelerated aging to simulate normal aging over 40 years under combined radiation and thermal stress conditions, chemithermal measurements, and analysis and reduction of data.

In the conduct of Task 2, accelerated aging protocols for cables were developed in the context of a rectangular matrix approach, called the "qualification aging matrix (QAM)," and were patterned after a good deal of successful research conducted previously. Twelve condition points in the QAM correspond to material conditions ranging from unaged to fully aged caused by multiple stresses throughout a full normal life in-situ. In a fully aged condition, operational integrity of these materials is expected to be intact throughout a postulated loss of coolant accident (LOCA). Plans were to analyze all condition points for a given material by chemithermal analytical methods. These methods include oxidation induction time (OIT), which measures product stability at high temperature and remaining antioxidant content (hence, remaining life); oxidation-induction temperature (OITP), which is a more rapid measure of stability and structural changes, related to OIT; and TGA, which measures thermal stability and gives some compositional information. Analytical methods for OIT and OITP tests had previously been standardized by Veridian PSR through funded government research. Standardization of these procedures involved multi-parametric experiments involving popular cable materials then under consideration. Sample mass, particle size, experimental conditions, and analyzer operating parameters and programs were all systematically investigated.

Similar experiments were conducted in this research to optimize TGA methodology. Critical cable-insulation products were identified through extensive contacts with material suppliers and users. Identification of critical cables for research is an on-going challenge, since installed-product operating data is continually updated, the number of manufacturers declines, virgin supplies of popular cables dwindle, and new products used for selective plant cable replacements become increasingly important. An array of ethylene-propylene rubber (EPR) and cross-linked polyethylene (XLPE) insulated cable products was selected, requested, and gradually amassed for this program. As these materials were being acquired, accelerated thermal aging procedures were being conducted, and chemithermal measurements of unaged and thermally aged samples were made. When enough materials had been thermally aged, gamma-irradiations were contracted for at the University of Maryland. To date, measurements of OIT, OITP, and TGA have been in line with expectations, but with some notable new discoveries, one of which is mentioned below.

The bulk of Phase 2 efforts focused on the application of chemithermal assay methods for new organic materials for o-rings and lubricants. This was the first work of its kind with these new materials. The o-ring materials studied were Nitrile-70, Nitrile-75, Butyl-70, and Ethylene Propylene-75. The lubricant materials were Krytox GPL-107 and Fomblin YR1800. OIT and OITP measurements were conducted on both aged and unaged materials in an experimental matrix that emulated the previous polymer cable measurements. The following observations were noted:

(1) The experimental aging and chemithermal assay matrix first engineered for the polymer cable materials proved to be a viable means of conducting degradation studies on other organic materials.

(2) OIT and OITP measurements conducted on these materials demonstrated the feasibility of using innovative new methods to characterize the oxidative degradation for o-rings and lubricants.

(3) TGA methods, while good for the o-rings, were not applicable to lubricants because of the material properties.

All o-ring and lubricant measurements have now been added to the comprehensive chemithermal assay measurement database.

This was successfully completed in June 2002 with the following major results:

- Establishment of a material-condition-measurement database consisting of over 1,100 OIT, OITP, and TGA measurements on 20 different types of cables, o-rings, seals, and lubricants
- A proven methodology for using OIT and OITP measurements on diverse materials that converges best practices from multiple research efforts
- A correlation study that shows the potential efficacy of correlating the following pairs of chemithermal measurement technologies: OIT-to-OITP; OIT-to-TGA; and OITP-to-TGA

The detailed results of this study are contained in the 132-page technical report entitled, Innovative Chemithermal Techniques for Verifying Hydrocarbon Integrity in Nuclear Safety Materials-Final Report (August 2002).

Planned Activities

The NERI project has been completed.

Nuclear Energy Research Initiative

Modular and Full Size Simplified Boiling Water Reactor Design with Fully Passive Safety Systems

Primary Investigator: Mamoru Ishii, Purdue University

Collaborators: Brookhaven National Laboratory

Project Number: 99-097

Project Start Date: August 1999

Project End Date: January 2003

Research Objective

The primary goal of this research project is the scientific design of a compact modular 200 MWe and a full size 1200 MWe, simplified boiling water reactors (SBWR). Specific objectives of this research are to:

- Perform scientific designs of the core neutronics and core thermal-hydraulics for small capacity and full size simplified boiling water reactors;
- Develop passive safety system design;
- Improve and validate safety analysis code;
- Demonstrate experimentally and analytically all design functions of safety systems for design basis accident (DBA); and
- Develop the final scientific design of both SBWR systems, SBWR-200 and SBWR-1200.

Research Progress

The following activities have been accomplished.

- Designs were developed for the compact SBWR-200 and SBWR-1200 thermal-hydraulic system and neutronic systems. The designs are shown in Figures 1 and 2, respectively, for SBWR-200 and SBWR-1200 reactors. Development of the designs involved identification of the principal design criteria dictated by the safe operation of the reactor, identification of coolant requirements, design of the engineered safety systems, and design of emergency-cooling systems based on passive systems and scaling analyses.
- A novel passive design of the hydraulic vacuum breaker check valve (HBVC) was developed and evaluated through RELAP5 simulation. This new check valve is based on the hydrostatic head, and has no moving components. It is comprised of only one additional tank, and one set of piping to the SP and DW. This system is simple in design and hence, easy to maintain and qualify for operation. The RELAP5 simulations were performed for the SBWR-200 and SBWR-1200, with the old mechanical and new HVBC valve, for the main steam line break (MSLB) transient. The new valve performance agreed quite well with that of the mechanical vacuum breaker check (MVBC) valve.

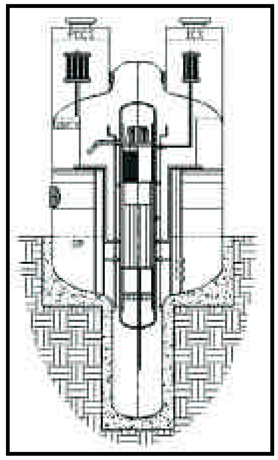


Figure 1. The schematic depicts SBWR-200 reactor containment.

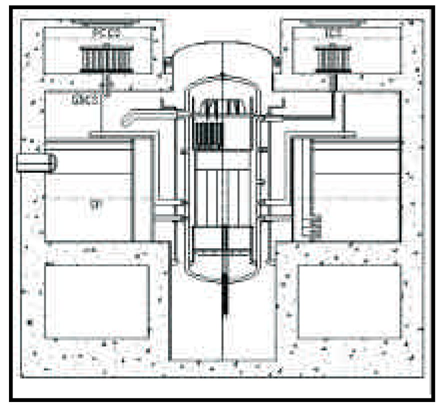


Figure 2. The schematic depicts SBWR-1200 reactor containment.

- The HVBC valve was designed, constructed, and installed for testing in the PUMA facility. Integral tests were carried out on the PUMA with the new HVBC valve for the MSLB and bottom drain line break (BDLB) accidents. The test data for the HVBC valve were compared with the data from MVBC valve, and the agreement was very good.
- A safety analysis for an anticipated transient with scram was performed for the SBWR-1200 using RELAP5. The transient considered was a main steam isolation valve closure accident with scram. The

simulation showed that the reactor is shut down without emergency water injection, and the decay heat is adequately removed by the ICS.

- Design basis accident scenarios were studied for the safety assessment of the SBWR-1200. Large break and small break loss of coolant accident (LOCA) integral tests for the SBWR-1200 were carried out in a PUMA integral test facility. These integral tests were perfomed to assess the safety systems and the response of the emergency core cooling systems to a LOCA.
- RELAP5/MOD3 best estimate reactor thermal hydraulic code was used to model the PUMA MSLB and BDLB integral tests. The analysis was used to demonstrate the safety features of the modular SBWR design and to validate the code applicability in the facility scope. Overall, the code gave a reasonably accurate prediction of the system thermal hydraulic behaviors. This allowed an accurate assessment of the design feature of SBWR-200 and SBWR-1200 safety components. It also indicated some code deficiency that should be improved for a better simulation.
- A detailed analysis and core design was performed for SBWR-200 and SBWR-1200. The neutronics work was performed in order to (1) acquire and validate the computer codes required for the neutronics design and analysis of the SBWR (HELIOS, PARCS, and RELAP5/TRAC); (2) develop neutronics and thermal-hydraulics models of the SBWR-600 and compare the results to the RAMONA-4B predictions; and to (3) perform designs of the SBWR-200 and SBWR-1200. Core depletion calculations were performed with PARCS, for a full fuel cycle analysis. A fuel lattice design was developed to optimize the fuel cycle safety parameters. A detailed study was carried out to improve the neutronics/thermal-hydraulics of all of the SBWR models. A fuel cycle analysis of the SBWR-200 and SBWR-1200 was also carried out.
- A stability study of the SBWR-600 (GE design) SBWR-200 and SBWR-1200 under normal startup and abnormal startup has been performed. Neither the geysering instability nor the loop type instability was predicted for SBWR-200 and SBWR-1200 by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillation was not observed at all because the power level used in the simulation was not high enough. A study was made of the potential instability by imposing an unrealistically high power ramp, with pressure restricted to 1.9 bar, as suggested by GE. Core flow oscillations of small amplitude were predicted by RAMONA-4B with a period of between 31.8 and 46.7 seconds, similar to that of the TRACG prediction by GE.

Planned Activities

The NERI project has been completed. Future plans include testing the performance of the final scientific design of the compact modular SBWR-200 and full size SBWR-1200.

NUCLEAR ENERGY RESEARCH INITIATIVE

A New Paradigm for Automatic Development of Highly Reliable Control Architectures for Future Nuclear Plants

Primary Investigator: Richard Wood, Oak Ridge National Laboratory

Collaborators: North Carolina State University; University of Tennessee

Project Number: 99-119

Project Start Date: August 1999

Project End Date: September 2002

Research Objective

This research focuses on development of methods for automated generation of control systems that can be traced directly to design requirements for the life of the plant. The final goal is to "capture" the design requirements inside a "control engine" during the design phase. This control engine is not only capable of automatically designing the initial implementation of the control system, but it also can confirm that the original design requirements are still met during the life of the plant as conditions change. Thus, the control implementation approach can provide a self-maintenance capability.

The control engine captures the high-level requirements and stress factors that the control system must survive (e.g., a list of transients, or a requirement to withstand a single failure). Therefore, the control engine is able to generate automatically the control system algorithms and parameters that optimize a design goal and satisfy all requirements. As conditions change during the life of the plant (e.g., component degradation, or subsystem failures) the control engine automatically "flags" that a requirement is not satisfied, and it can even suggest a modified configuration that would satisfy it.

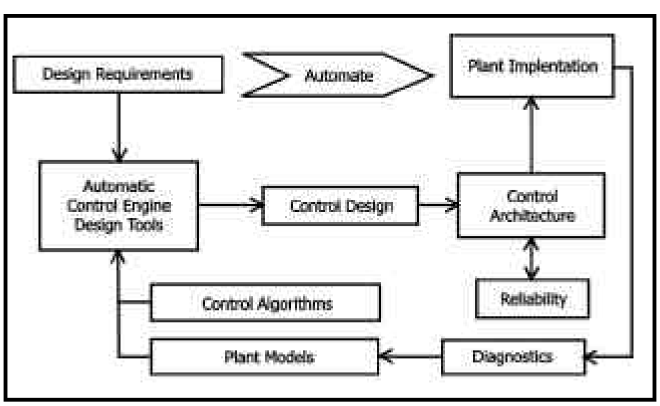


Figure 1. The schematic diagram shows the automated control design process.

This control engine concept is shown schematically in Figure 1.

Research Progress

Project accomplishments include:

Advanced Control Tools and Methods: Research in this area is directed toward developing and demonstrating the control engine concept. Libraries of control algorithms have been developed, with a selected set having been demonstrated for prototypical control engine problems. The automatic control engine has been successfully applied to the generation of controllers for U-tube steam generators (UTSGs) and feedwater systems using a full plant simulation of a pressurized water reactor (PWR). In addition, the self-maintenance capability of the control engine concept has been demonstrated using fault injection with the simulated plant. In the application, a multi-step process was used. First, diagnostic modules identify a sensor failure. Next, the control system simulation model is updated to reflect current plant status. Subsequently, the control engine automatically validates the control design against the system requirements and constraints and, for this application, generates a new control solution. Finally, the new controller software is implemented and permits continued normal operation in spite of the degraded plant conditions. As additional research under this area, general methodologies have been developed for control priority mode selection and for handling the sensor and actuator nonlinearities as piecewise linear functions.

Advanced Monitoring and Diagnostics: The objective of this task is to develop an on-line monitoring system for fault detection and isolation of sensors and field devices in a nuclear power plant. Data-driven models have been developed for the characterization of sub-system dynamics for prediction of state variables, control functions, and expected control actions. The fault detection and isolation (FDI) modules combine system operational knowledge (including system simulation) and a rule-based logic (with options for other fault pattern classification techniques) for identification of both single and dual faults in dissimilar sensor and field devices. The FDI techniques have been successfully applied to a laboratory process control loop and a UTSG using the full-plant PWR simulator. Demonstration of the FDI module for both single and simultaneous dual faults has been accomplished and includes the following highlights:

- Rule-based decision making
- Fault isolation using fault residuals and pattern classification
- Steady state and transient plant operation conditions
- Combination of sensors and valve actuators
- Demonstration of the functional features of the FDI module using the PICASSO interface system from the Halden Reactor Project

Nuclear Power Simulation and Reliability Methods: Research in this area involves development and application of a full plant engineering simulation code to represent the dynamic response of PWRs during normal operational transients as well as design basis events. Work under this task required the addition of a full balance-of-plant model, as well as other improvements, to the plant simulator. Options have been added to the code to allow for fault injection such as degradation in the heat transfer across the steam generator from both fouling and blocked or plugged tubes, as well as the corruption of sensor outputs through step and ramp changes in sensor output with arbitrary levels of random noise. This simulator serves as the demonstration platform for the control and diagnostic methods developed in this project. Additionally, as an element of the control system design approach, a strategy and methods have been defined to integrate and automate instrumentation and control (I&C) system reliability analysis with nuclear power plant simulation. To realize this capability, the control system simulation is structured as a reliability block diagram model representing all units that compose an I&C system. For each unit, input data consists of the type, tag number, power sources, applicable failure modes (high, low, on, of, open, closed, etc.), failure rates, linkage to all of the unit input sources, and linkage to all of the unit output destinations. Using this simulation capability, a systematic evaluation of the effect of each failure mode can be accomplished and a comprehensive failure modes and effects analysis established. A prototypic application of the basic elements for this approach was performed using representative feedwater control system architectures.

Nuclear Information System Architecture and Integration: Research addressing the control and information system architecture for future nuclear power plants involves the evolution of the Plant-Control Computing Environment (PCCE) concept. Functional requirements for the PCCE have been develop to address general design attributes, human-system interface requirements, control application interface requirements, computing platform interface requirements, monitoring and control requirements, fault handling and recovery requirements, system management requirements, and configuration requirements. A prototypical PCCE has been developed to investigate implementation issues and to serve as the framework for an integrated demonstration of the research products for this project. The PCCE provides a distributed computing environment supporting a high-level supervisory control and monitoring system (see Figure 2). Application programming interface elements have been developed to

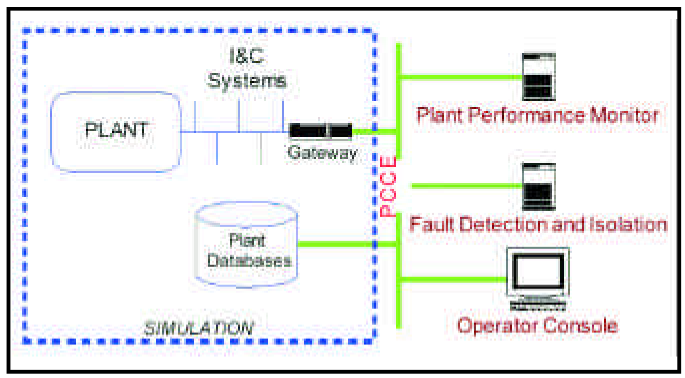


Figure 2. The graphic illustrates the plant-control computing environment and applications.

provide communication and information services for the other research modules (e.g., Fault Detection and Isolation, Plant Performance Monitor, PICASSO-based Graphical User Interface, Full-plant PWR simulator). An integrated demonstration of the research products from this project was performed using the PCCE as the interface platform. The prototypic application illustrates the interaction among the control, diagnostic, simulation, and interface elements. In addition, the self-maintenance capability of the control system architecture was demonstrated through injection of faults into the plant simulation, detection of degraded conditions by FDI modules, adaptation of the control system simulation model to reflect the current plant status, automatic assessment of the control system, and regeneration of the control solution to accommodate changes in the plant condition.

Planned Activities

In Phases 3, an integrated demonstration of the research products from this project will be accomplished. The requirements-driven control system design will be investigated further through additional applications of the automatic control engine concept. On-line diagnostics system requirements will be determined through stand-alone and integrated (i.e., coupled control and diagnostics) implementation of the FDI modules. The automated reliability analysis methods will be developed through an application to a feedwater control system architecture.

NUCLEAR ENERGY RESEARCH INITIATIVE

Multi-Application Small Light Water Reactor (MASLWR)

Primary Investigator: S. Michael Modro, Idaho National Engineering and Environmental Laboratory

Collaborators: NEXANT; Oregon State University (OSU)

Project Number: 99-129

Project Start Date: October 1999

Project End Date: December 2002

Research Objectives

The primary project objectives are to develop the conceptual design for a safe and economic plant and to test the design feasibility. A small, natural-circulation, light water reactor is proposed with the primary goal of producing electric power, but including the flexibility to be used in process heat applications with deployment in a variety of locations. Economic and engineering analyses will be used to address the design and safety attributes of the concept. These analyses will be coupled with testing in an integral test facility to demonstrate the concept's technical feasibility.

Research Progress

Three major efforts were addressed in the first year of the project:

- To establish the requirements and design criteria applicable to the MASLWR
- To develop a baseline design concept, including a preliminary cost-estimate
- To generate the general scaling methodology needed to construct an experimental facility with which to test significant features of the baseline concept

The initial concept, explored during Year 1 activities, was a natural-circulation design to be operated at approximately 1,000 MWt and 5.4 MPa steam pressure. This design included four horizontal U-tube type steam generators located at a height of 36 meters above the thermal center of the reactor core. A cylindrical containment, 30 meters in diameter, housed the reactor and primary system and the required support systems and equipment. The preliminary estimates for this design indicated that the busbar cost would be about \$0.057/kWh. It was concluded that if the basic concept principles identified at the outset of the project were maintained (i.e., a pressurized water system with natural circulation), cost reduction could be achieved only by using smaller, simpler, factory-assembled units.

Consequently, activities in Years 2 and 3 have focused on developing a modular reactor design that consists of a self-contained reactor vessel assembly, steam generators, and containment. These modular units would be manufactured at a single centralized facility; transported by rail, road, and/or ship; and installed as a series of self-contained units.

Design optimization studies yielded a natural-circulation concept with a helical-tube steam generator, shown in Figure 1. Primary-side fluid flowed through the shell of the steam generator, and the secondary-side fluid was inside the tubes. The advantages of this system include low primary-side frictional losses, and the use of a single pressure vessel that encloses the reactor core and the steam generator. The primary pressure vessel is located within a containment that is a vertical, cylindrical vessel with elliptical heads. The primary vessel is partially submerged in liquid. The containment itself is submersed

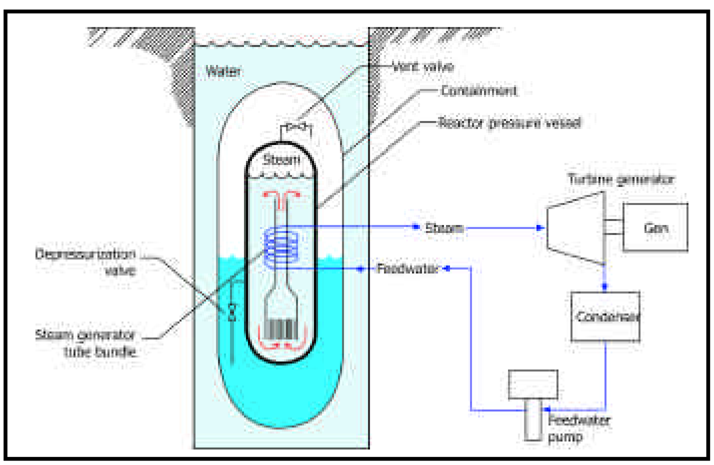


Figure 1. The schematic illustrates the natural-circulation, helical-tube steam generator of the MASLWR Baseline Design Concept.

in a below-grade pool of water, which serves as a passive ultimate heat sink.

A RELAP5 model was developed for the MASLWR to determine the operating characteristics of the design and to perform accident studies. Performance characterization studies demonstrate that the modular, natural-circulation design will operate at approximately 150 MWt and 1.5 MPa of steam pressure, which yields 35 Mwe at a thermal efficiency of 23 percent.

Safety analysis studies were performed for hypothetical accident scenarios based upon Final Safety Analysis Report Chapter 15 Guidelines for current generation pressurized water reactors, and grouped by frequency of occurrence. All the analysis cases demonstrate acceptable performance with significant margins to safety limits.

A MASLWR baseline plant consisting of 30 power generation units was used to develop schedule and cost estimates. The construction and startup schedule is estimated at 18 months for the first two power generating units, and at 36 months for the entire baseline plant. Total capital cost is estimated at $1,200/kWe and the busbar cost is $0.034/kWh.

Plant arrangement drawings have been developed for the 30-unit power generation complex and also for a six-unit plant. These drawings include the reactor and turbine generator buildings and a fuel handling and maintenance building. Other major facilities are also provided, including a main heat rejection system, control building, remote shutdown building, machine shop, and warehouse, as well as facilities for personnel services, plant services, administration and training, waste treatment, and guardhouses.

A preliminary concept was also developed for a stand-alone, single-unit plant, which would be capable of operation for a total of 60 years, consisting of two 30-year, self-reliant periods, with a core designed with a fuel cycle life of 5 to 10 years. Three to six refueling outages will be necessary during each self-reliant period. All new and spent fuel for the period will be stored onsite, and will be shipped offsite for disposal or reprocessing at the end of the period.

Seawater desalination in combination with power generation was also investigated for the 30-unit plant. The following seawater desalination processes were considered:

- Multistage flash distillation
- Multi-effect distillation
- Reverse osmosis

The results showed that reverse osmosis is the most attractive alternative. It reduces the electrical power output to 25.4 MW, and produces 15.4 MGD of desalinated water. The additional capital cost is $62.2 million and the annual power cost is $4.0 million.

District heating and/or cooling was also investigated. However, assuming a minimum practical hot water temperature of 180ºF, the turbine exhaust pressure would need to be 10 psia compared to 0.75 psia for the baseline design, thereby reducing the electric power capability from 35 MWe 18 MWe. A power reduction of this magnitude for using turbine heat exhaust for district heating and/or cooling makes the economics questionable.

A fuel handling and maintenance concept has been developed for the MASLWR. The facility will include one out-of-service module, which will be available as an immediate replacement for a module requiring refueling, thereby minimizing outage time. The modules will be transported intact to and from the refueling/maintenance facility, entirely under water. Details of the disassembly and reassembly processes have been developed. The refueling and maintenance activities are entirely automated.

The overall scaling analysis has been completed for the MASLWR test facility at OSU. It includes a Natural-Circulation Scaling Analysis, a Sump-Recirculation Scaling Analysis, a Reactor Coolant System Depressurization Scaling Analysis, and a Containment Pressurization Scaling Analysis. The analysis indicates that full power steady-state operations will be well-simulated in the test facility. The facility will obtain valuable information on helical tube steam generator performance. The power-to-helical tube-surface area scale ratio has been preserved. The unique OSU containment design preserves both the power-to-containment volume scaling and the power-to-active heat transfer surface area scaling requirements. The Automatic Depressurization System (ADS) Blowdown behavior and containment response will be well simulated in the test facility. The thick-wall vessel required to operate the facility at full pressure (1.25 in of thickness) results in too much reactor pressure vessel mass. This will likely prolong the downcomer hot-wall effect during long term sump recirculation cooling period. The hot wall effect will lead to conservative results with regards to core cooling during this period.

The GOTHIC computer code was used to determine the maximum design pressure for the test facility containment. The results were generally consistent with the RELAP5 calculations for steam vent actuation without submerged ADS operation.

Design specifications for manufacture were developed for the test facility containment vessel and its associated liquid pool based on the Containment Pressurization Scaling Analysis.

The MASLWR test facility has been completed mechanically. The containment and liquid pool vessels have been delivered and the interconnecting piping has been connected to the test loop.

Figure 2 shows a photograph of the MASLWR test facility at OSU. All of the primary loop hardware, controls, instrumentation, and related software have been purchased. Also, the primary loop of the facility and all of the supporting structure have been set in place, and

Figure 2. The primary loop and structural support of the MASLWR Test Facility have been constructed at Oregon State University.

power and controls have been installed. The instrumentation has been installed and its operation verified for reading by the Data Acquisition System. Initial measurements of the test facility's component volumes have been obtained and primary loop pressure drop measurements have been preformed. Cold Shakedown tests were performed and were repeated following replacement of a leaking hot leg riser seal.

Additional work is being performed to assess the GOTHIC condensation models and 3-D nodalization. The test facility input deck for GOTHIC has been modified to incorporate the features of the final containment design.

Programming of the instrumentation and control software for the test facility is complete. The control systems will allow OSU staff to perform testing from a separate control room. Control algorithms have been developed, and are ready for testing.

Three comprehensive papers have been peer-reviewed and prepared for presentation.

Planned Activities

A series of Hot Shakedown tests are planned and are in progress. Phase 1 will include a series of steady state single-phase natural circulation tests to characterize the operation of the facility during single-phase conditions. The test will measure flow and power enhancement effects due to subcooled boiling, and to parameterize the performance of the helical tube steam generator as a function of feedwater flow rate and core power. The steady-state control algorithms will be be tested concurrent with the hot shakedown tests.

Phase 2 testing will consist of a several transient blowdown tests, which will include the test facility containment vessel and associated liquid pool. These tests are still being designed.

Two heater rods failed during performance of Phase 1 hot shakedown tests. The heater rod design is being analyzed to see if the two failures are related to the heater design. Current plans call for hot shakedown testing to recommence after repair of the leaky heater rod seal.

Additional work is being performed to assess the GOTHIC condensation models and 3-D nodalization. The test facility input deck for GOTHIC has been modified to incorporate the features of the final containment design.

NUCLEAR ENERGY RESEARCH INITIATIVE

STAR: The Secure Transportable Autonomous Reactor System, Encapsulated Fission Heat-Source (ENHS Project)

Primary Investigator: Ehud Greenspan, University of California, Berkeley

Collaborators: Argonne National Laboratory (ANL); Lawrence Livermore National Laboratory (LLNL); Westinghouse Electric Company LLC

Project Number: 99-154

Project Start Date: August 1999

Project End Date: December 2002

Research Objectives

The primary objectives of the ENHS study were to assess the feasibility of

(1) Designing lead-bismuth or lead cooled reactor cores for 20 years of full power operation without refueling and with nearly zero burnup reactivity swing, using either Pu-U or enriched uranium fuel;

(2) Designing the ENHS to have 100 percent natural circulation in both primary and intermediate coolant systems with reasonable module dimensions and weight;

(3) Designing the ENHS module to be free of mechanical connections to the energy conversion system;

(4) Designing an intermediate heat exchanger (IHX) that is an integral part of the ENHS module vessel and assessing the feasibility of manufacturing it;

(5) Fueling the ENHS module in the factory, transporting it fueled and weld sealed to the nuclear power plant site and installing it in the reactor pool;

(6) Removing the module from the reactor pool at end of life and transporting it unopened to a recycle facility;

(7) Designing the ENHS to have autonomous load-following capability;

(8) Designing the ENHS reactor to have passive safety so that postulated accidents will not damage reactor components;

(9) Attaining a high energy-conversion efficiency;

(10) Assessing the feasibility of using alternative coolants while retaining the unique characteristics of the ENHS; and

(11) Assessing the ENHS reactor ability to meet the goals set for Generation-IV reactors.

Research Progress

The design domain has been identified for cores that can operate at 125 MWth for 20 years with nearly zero burnup reactivity swing. The core design variables include the core height, lattice pitch-to-diameter ratio, and fissile fuel contents. It was determined that it is possible to design such cores using either Pu-U fuel having approximately 11-12 weight percent Pu or uranium enriched to approximately 13 weight percent, both in alloy with Zr (10 weight percent). The core life is limited by radiation damage to the clad. A fission gas plenum volume that is equal to the fuel volume is required to accommodate the fission gas pressure buildup and maintain the clad integrity. The core designs are simple: they have uniform composition, no blanket or reflector assemblies, and a single central safety assembly. Six axially movable absorber assemblies surrounding the core are used for reactivity control. They are lifted almost completely above the fuel level to bring the core to full power. A combination of tungsten and B_4C is used for the absorbing material; its specific density is larger than that of Pb-Bi so that scramming can be done by gravity. The maximum change in k_{eff} throughout 20 years of full power operation is only 0.2 percent or about $0.5. After the reactor is brought to full power it may be necessary to adjust the peripheral absorber elevation approximately once every two years. The core power shape and reactivity coefficients stay exceptionally constant throughout the life-cycle. Figure 1 illustrates the extent of radial power shape variation in 20 years.

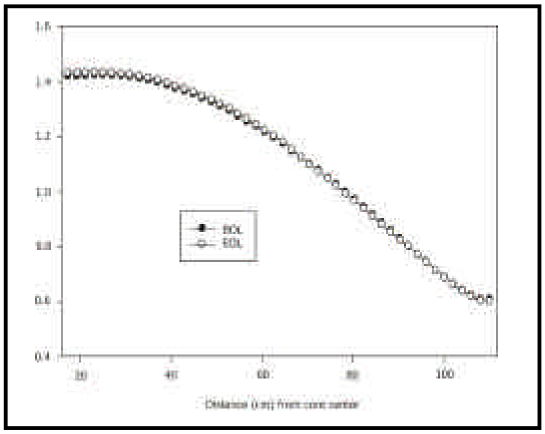


Figure 1. The graph shows axially integrated radial power distribution variation over 20 years of full power operation of the ENHS module.

It was found feasible to design ENHS modules to deliver 125 MWth from the primary to the secondary coolant through a 4-mm-thick, intermediate heat exchanger with a maximum primary coolant to maximum secondary coolant temperature drop of 50°C and with 100 percent natural circulation. The required reactor vessel height is close to 20 meters. In fact, good synergism was found between the requirement for 100 percent natural circulation and the requirement for a primary-to-secondary temperature drop of 50°C. It was also found that the ENHS reactor can maintain 100 percent natural circulation over the entire range from nominal power to decay heat power level.

A significantly more compact and lower weight ENHS module has been conceived and partially analyzed. It uses cover-gas lift-pumps to enhance coolant circulation. The gas circulators are located outside of the reactor vessel. With use of a lift pump, the ENHS module is approximately 10 meters long—about half the length and volume of the original module, with 100 percent natural circulation. The power level of an ENHS module with a lift pump could be approximately 190 MWth.

An IHX that is integrated within the ENHS module vessel walls and has practically identical heat transfer characteristics to those of the originally proposed confinement wall has been conceived and analyzed. It consists of rectangular channels circumscribing the primary coolant riser in the space between the inner and outer structural walls. This integrated IHX was found able to withstand the loads that are expected during transportation, installation, and operation. Relative to circular tube IHX, the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid spacers. Another innovative IHX design has been conceived; it consists of nested tube bundles. Advantages of the proposed nested channels IHX concept are that: it can be made from off-the-shelf components; it has high rigidity against buckling; and it is easier to fabricate.

A strategy was developed for fueling the ENHS module in a factory and transporting it to the nuclear power plant site as a weld-sealed unit. To avoid fuel damage during transport and module installation, the fuel is embedded in solidified Pb-Bi that fills the vessel to above the fuel rods. The total weight of the reference ENHS module for transportation is estimated to be about 300 tons. This is approximately half the weight of steam generators for large pressurized water reactors (PWRs) that have been transported from the factory to nuclear power plant sites. Dimensions of the reference ENHS module are somewhat smaller than dimensions of a large PWR steam generator. It appears feasible to factory manufacture and fuel the ENHS module and ship it to the site using available transportation equipment.

At the site, the ENHS module will be inserted into the pre-heated secondary Pb-Bi pool vessel. Hot Pb-Bi is then pumped into the ENHS vessel through a pipe. This hot Pb-Bi, along with the hot Pb-Bi in the pool, will melt the solid Pb-Bi at the lower part of the vessel. It has been found feasible to melt the primary coolant and bring it to the reactor startup temperature of 350°C within approximately two days of its insertion to the pool without having to use special external heaters in the module.

After 20 effective full power years, the ENHS module will be removed from the reactor pool into a storage vault on site. Before removal, forced cooling will be applied and the Pb-Bi will be pumped out the ENHS vessel until it reaches the upper level of the fuel rods. The rest of the Pb-Bi will be replaced by Pb. The Pb will be solidified approximately 10 days after shutdown by cooling the outer walls of the vessel and the surface of the shutdown assembly channel located at the center of the core. If cooling is provided only at the outer boundary, the Pb can be solidified after ~22 days. The ENHS module will stay in the storage vault until the decay heat drops to a level such that passive cooling will permit the Pb to remain solid during transportation. The ENHS with the solidified Pb will then be prepared as a licensed shipping package and transported to a recycling facility. Two packaging configurations were conceptually designed for shipping the used ENHS module. The first uses a cask to contain the ENHS module and is based on conventional Type B spent fuel shipping casks. The second configuration uses the ENHS module as the containment system and is based on the decommissioned Trojan and Shippingport reactor packages.

Due to its negative temperature feedback and to cooling based on natural circulation, the ENHS reactor has a load-following capability over a wide range of power. The reactor power level will adjust itself to the power demand without operator intervention. This autonomous load following capability has been demonstrated in numerical simulation. The operators need only perform a startup and a periodic surveillance function. Both functions may potentially be accomplished remotely from a centralized facility that services many units.

A preliminary safety analysis was performed for the reference ENHS design. The accidents considered so far include a startup accident, a loss of heat sink accident, and a steam line break without scram accident. It was found that under all accident conditions considered, fuel and clad temperatures will remain significantly below the safety limits and the integrity of all systems will be maintained. Figure 2 shows that in a loss of heat sink without scram accident-the worst plausible accident identified-the peak fuel temperature will not exceed its steady state operating conditions. The decay heat is passively removed by the reactor vessel air-cooling system (RVACS). Contributing to the exceptional safety features of the ENHS are the lack of pumps and valves in the primary and intermediate cooling systems, use of full natural circulation, high heat capacity, low power density, more than a 1,000°C margin between the coolant operating and boiling temperatures, and availability of very small excess reactivity for positive reactivity insertion accidents.

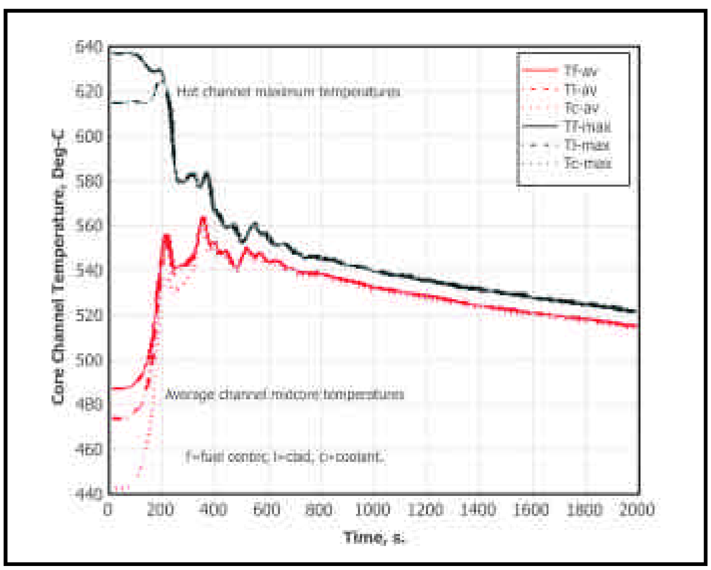


Figure 2. The plots show temperature evolution in selected core locations following a loss of heat sink accident without scram.

Steam generators have been designed to meet several unique requirements that are dictated by the ENHS reactor layout:

- Effective utilization of the pool volume surrounding the ENHS module
- Minimum friction losses so as to enable 100 percent natural circulation of the intermediate coolant
- Absence of a mechanical connection with the ENHS module
- Minimum flow rate of water into the intermediate coolant pool in case of a breach in steam generator tube or failure of other water-containing component
- Accommodation of a large thermal expansion
- Ease of inspection and maintenance
- Modular design that is easy to install and replace

Two-dimensional flow simulation showed that at nominal operating conditions only 3 percent of the intermediate coolant flow bypasses the steam generators. This small bypass has a negligible effect on the ENHS energy balance. The 2-D flow simulation also revealed that there is a stagnation zone at the bottom of the reactor pool. A design modification that eliminates the stagnation zone has been worked out; it involves insertion of a vertical partition in the reactor pool. Another novel plate type steam generator has recently been conceived that may allow making the reactor pool more compact or integrating the steam generator with the IHX.

A Rankine steam cycle was designed to match the heat source characteristics of the ENHS steam generators. The gross thermodynamic efficiency for converting the ENHS heat to electricity was calculated to be 40 percent; the corresponding net efficiency is 38 percent. These efficiency values correspond to a simple energy conversion system that has no reheat and moderate steam pressure of 120 bars. The main effect of the reheat is a reduction of the problematic low-pressure steam moisture. Only marginal power addition is achieved—less than 1 percent. Increasing the steam pressure to 160 bars can increase the net efficiency to 40 percent. An even higher efficiency can be attained using supercritical steam. However, the steam moisture level increases with steam pressure increase, unless reheaters are used. An energy conversion system using CO_2 working fluid appears to make a better match with the ENHS than a steam driven energy conversion system. The ENHS reactor can be designed to provide, in addition to electricity, process heat for desalination, for district heating as well as for industrial applications.

The neutronic and thermal hydraulic characteristics of Pb-Bi and Pb are very similar. The same ENHS reactor design can, in principle, operate with either one of those coolants. ENHS can also be designed to provide nearly zero burnup reactivity swing and 100 percent natural circulation using sodium for the primary coolant and possibly also for the secondary coolant. The sodium-cooled designs feature more compact smaller diameter cores but larger core peak to average power ratio and smaller thermal margins.

The study found that the ENHS reactor concept can meet very well all of the goals set for Generation IV reactors that have been examined, including the following:

(1) Highly sustainable energy supply. The ENHS maintains its fissile fuel inventory constant; all that is needed for reusing the fuel discharged from one module is to remove fission products, add depleted uranium and re-fabricate fuel rods.

(2) Low waste. The TRU can be recycled many times.

(3) Extremely high level of proliferation resistance. There is no access to the fuel in the host country and no access to neutrons. The host country gets energy security with no need to invest in fuel cycle technologies.

(4) Superb safety and reliability. The reasons for these features were discussed in the previous page. As a consequence of these features there is no need for emergency planning zone outside of ENHS nuclear power plant fence.

(5) Very low risk to capital. This is due to relatively low cost per module, standard design with factory assembly line fabrication, short construction time, superb safety, and large tolerance to human errors. However, the economic viability of the ENHS has not yet been examined since it was out of the scope of work. Nevertheless, general considerations and very preliminary examination of the ENHS fuel cycle cost indicate that there is a good probability that the ENHS will be economically viable despite its small unit size.

Therefore, it is recommended that the ENHS reactor concept be considered as a candidate for Generation IV reactors. A follow-on study to assess the economic viability of the ENHS reactor is suggested. Before embarking upon such an assessment, the recommendation is made to

- Define the logistics and infrastructure required for fabricating, fueling, transporting, operating and handling of the ENHS module;
- Estimate the market potential for ENHS reactors;
- Assess the feasibility of a number of design variants for the ENHS along with the maximum power and maximum discharge burnup the ENHS module can handle; and
- Work out optimized designs with greater design detail for subsystems of the ENHS reactor.

The ENHS is targeted primarily for a market of small turnkey nuclear power plants with full fuel cycle services. The potential market size for the ENHS was recently roughly estimated to be between 200 to 1,400 modules per year over the next 40 years.

More detailed description of the ENHS reactor concept and its feasibility study is documented in 35 publications.

Planned Activities

The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

On-line, Intelligent, Self-Diagnostic Monitoring for Next Generation Nuclear Power Plants

Primary Investigator: Leonard J. Bond, Pacific Northwest National Laboratory (PNNL)

Project Number: 99-168

Project Start Date: September 1999

Project End Date: January 2003

Research Objectives

The objectives of this effort are as follows:

- To focus the project on relevant Generation II, III, and IV components and reactor system aging degradation mechanisms
- To specify the experimental apparatus, configuration, and instrumentation, and to allow the selected components and degradation mechanisms to be fully exercised, measured, and analyzed
- To develop and codify a Self-Diagnostic Monitoring System (SDMS) computer architecture that is modular, allows a hierarchical diagnostic modeling approach, and is extensible to the inclusion of the full suite of degradation mechanisms; this extensibility to include all active and passive components, systems, and structures to be found in the next generation of U.S. power reactors
- To design, fabricate, and test advanced smart multi-sensor RF tag modules that will serve as the Level 3 nodes in the SDMS hierarchy; the Smart Multi-Sensor Tag (SMST) to provide for wireless data communication links between distributed intelligent sensors and the Level 2 (system level) processing nodes
- To develop the analytical methods and algorithms that will provide the diagnostic and prognostic processing to enable full condition-based operations and maintenance including first-principles life-cycle asset management; to develop a pragmatic solution needed for applications to real-world problems

Research Progress

All of the foregoing objectives have been achieved by this project. A discussion of progress in specific topical areas follows:

Wireless Technology: Smart Multi-Sensor Tag (SMST) units were designed, fabricated, and successfully tested to show that a highly reliable fault-tolerant wireless system could be constructed. A remote, totally portable data display showing system conditions and degradation alarms was demonstrated in the laboratory.

System Modification: A heat exchanger was purchased and installed in the system along with appropriate feedback controls to maintain system temperature within acceptable limits for extended filter testing. The unit is a shell and tube two-pass stainless steel U-tube assembly that uses laboratory building water as a heat sink. Performance was found to be ± 1°C in the process stream during testing.

Analysis Methodology: The analytical approach taken by the PNNL team was completed. Called condition-based operations and maintenance (CBO&M), this method is aimed at the immediate detection and diagnosis of off-normal equipment operation and the identification of the root cause stressor(s) responsible for this condition. This approach yields a computerized real-time picture of the process problem and promises a clear understanding of the temporal nature of the solution. With such a tool, true asset management can proceed using informed decisions based on known conditions, defined degradation rates and, in most cases, accurate estimates of equipment remaining life (prognostics).

The basic concept for CBO&M stressor-based analysis centers on the fact that by understanding the stressor characteristics, an anticipatory indicator is provided for mapping subsequent damage through the activation of a resulting degradation mechanism.

The premise of this methodology is that, by not trending a performance metric per se, but by focusing on trending the stressor characteristics, a precursive relationship can be derived that will allow a much more accurate projection of the remaining useful life. Figure 1

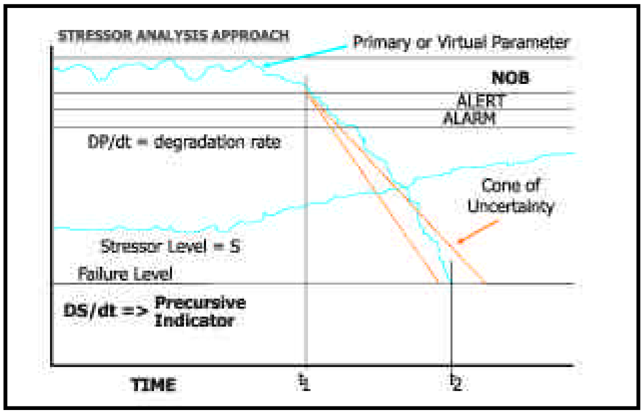
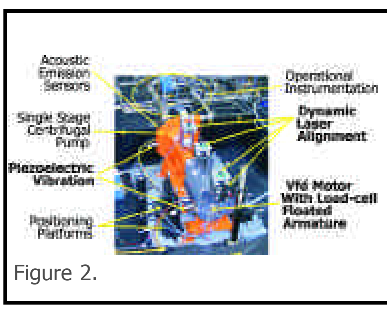


Figure 1. The graph shows the stressor measurement effect on prediction uncertainty

shows the expected result in narrowing the failure cone of uncertainty by keying on the stressor itself.

By monitoring the slope of the stressor intensity, researchers obtain a precursive operator feedback and a measure of the rate of change in the performance degradation. Thus, the stressor slope can be used to forecast and refine the path of the performance vector.

Pump Testing: The two most predominant mechanisms that result in centrifugal pump failure are cavitation and vibration. The instrumentation required to quantify stressor intensity and examine the physical effects of degradation have been developed for cavitation and vibration mechanisms in centrifugal pumps (Figure 2) and fouling in reverse osmosis heat exchangers (Figure 3).



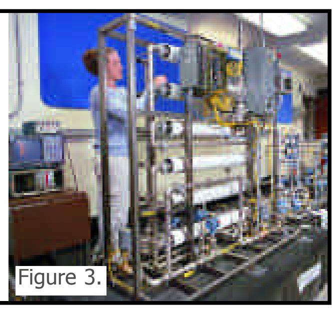


Figure 2. This photograph shows the cavitation-vibration instrumentation developed for this experiment.

Figure 3. The researcher is involved in reverse osmosis testing.

Vibration Mechanisms: High frequency pump and motor instrumentation was developed to measure the magnitude of the stressors and resulting reaction forces on the bearings. Three measurement systems were used:

(1) Standard vibrational instruments were placed on both the pump and motor to provide a common, well-accepted reference for vibration diagnosis of machine faults.

(2) A laser dynamic position sensing system for a pump-motor set misalignment was designed and built to provide carefully quantified angular and parallel offset and to measure time-dependent response.

(3) A load cell system was developed to characterize the effects of static and dynamic rotational stressors on the armature bearings.

Analyses of the resulting spectral peaks show an excellent correspondence between the (laser) motor position indication, the vibration response, and the dynamic force loading on the bearings. Orbital and harmonic motion of the pump and motor are clearly indicated and can be readily correlated through the speactral peaks of all three sensing systems. Laser motion spectra were actually found to correlate more cleanly to the peak structure of the load cell spectra than did either accelerometer vibration sensor. By driving a three-dimensional visualization program with position data from the laser device, a clear, intuitive understanding of the primary pump-motor oscillations and their associated harmonics was obtained.

By utilizing the discrete spectral signature produced by the bearing load cells, a direct correlation between angular misalignment and the reduction in bearing life was determined. A life factor equation of the form:

$$LF = \left(\frac{P_{ei}}{P_{ea}}\right)^{p}$$

was used to derive the stressor to life reduction factor of

$$LF = 1 - (0.02) \times [\text{angular offset}]$$

where the angular offset is specified in mils of base displacement of the test pump. While not in a generally usable form since this measurement is specific to the test apparatus used, it nevertheless shows the closed form equation relating the stressor to the useful residual life (URL) of the machine. This fulfills the project goal of generating a proof of principle correlation between the primary stressor (misalignment) and the equipment URL.

Cavitation Mechanisms: The initial goal for the cavitation test series was to characterize the operational data as well as the spatial and spectral nature of the cavitation produced in a single stage centrifugal pump. To this end, highly accurate operational instrumentation was used to measure the motor current, suction pressure, and temperature and the discharge pressure, temperature, and flow. Specialized acoustic sensors were then installed in the test pump per Figures 4a and 4b. These sensors were placed in direct contact with the pumped fluid to provide a clear view of the acoustic energy impacting the wall of the volute.

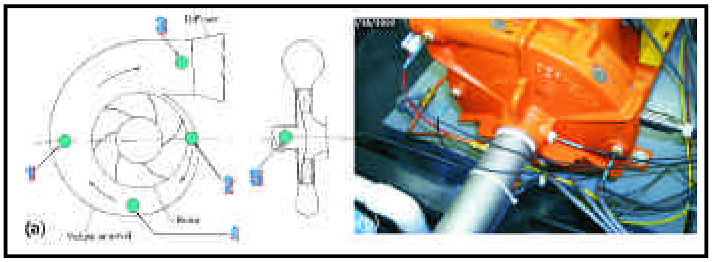


Figure 4. (a) The diagram illustrates acoustic sensor placement. (b) The photograph shows a view of the test pump.

After problems were encountered with electromagnetic interference, it was found necessary to remove the pump variable frequency drive (VFD). With the motor operating without VFD, a clear cavitation signature was obtained and signatures were obtained for both direct fluid contact and contact with the exterior surface of the pump volute. Highest values of signal-to-noise ratio were obtained using a non-intrusive probe located near the suction of the pump.

A series of tests were performed where the pump was run under conditions with varied suction pressures across the range from well above the manufacturer's suggested Net Positive Suction Head (NPSH) to the minimum suction pressure that the pump would produce. Acoustic data records were captured for each case and their respective spectra were produced using Fast Fourier Transform (FFT) processing. By normalizing the cavitation acoustic spectra to the baseline case (no cavitation), Figure 5 was the result.

Figure 5 shows the acoustic signal amplitude as a function of the suction pressure. A distinct inflection point is seen at 20 psia which has been labeled the incipient cavitation point. Above this suction pressure the literature suggests that the increasing signal is due to higher acoustic conduction through the single phase fluid. Below

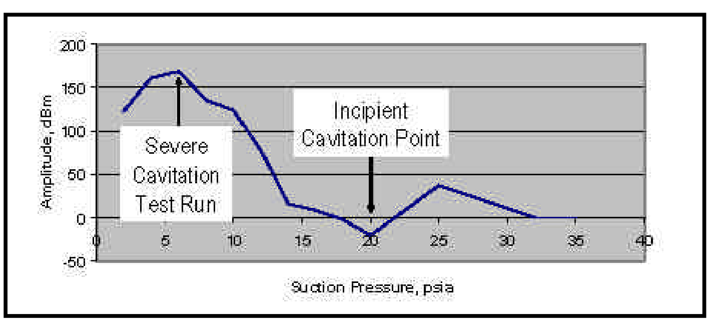


Figure 5. The graph is a cavitation test composite.

the incipient cavitation point, increased cavitation implosion is responsible for the dramatic increase in acoustic energy. It is interesting to note that the stated NPSH for the test pump equates to 18 psia, indicating the designers' lack of accuracy in determining the NPSH value, or perhaps their willingness to operate the unit under a mild cavitation condition. A maximum energy signal is seen at about 6 psia. It was speculated that these lower pressures produce a slug flow phenomena in the suction line that again reduces the acoustic signal. This hypothesis was supported by the observation of slug flow in the Lexan sections that were inserted into the suction pipe of the test rig. Figure 5 also suggests one method of scaling the intensity of the acoustic energy that results in metal removal.

A continuous cavitation run was initiated on September 2 and continued 24 hours a day for 4 weeks (with the exception of a 4-hour power outage in the laboratory). The test pump was then secured, drained, and disassembled to obtain wear readings relative to the baseline. With the exception of the wear ring clearances very little metal removal was observed. The impeller to volute gap in this area indicated a 10 mil increase in clearance.

Without performing further cavitation runs, only a simple linear correlation can be derived from the available two point data set. When combined with the acoustic intensity measurement this provides a "zeroth order" approximation to a correlation that relates suction pressure differential from incipient cavitation (the primary stressor) to the degradation rate of the pump. Making several major assumptions about the linearity of a logarithmic intensity scale and the validity of its relationship to metal removal rate, it is possible to derive an equation of the following form:

$$MRR = K\left[10\exp\left(13.9\, PSID_{NPSH}\right)\right]T(da)$$

Where:

- MRR is the metal removal rate
- T is the cavitation time in days
- $PSID_{NPSH}$ is the differential pressure between the operating point and the pump NPSH limit
- K is a material and geometric constant dependent on the specific pump
- The coefficient 13.9 is the slope of the (logarithmic) acoustic intensity line from Figure 5 and is in db/psid.

The task then remains to relate this to the useful residual life through an understanding of pump performance as a function of wear ring clearance. When the pump internal circulation reduces its throughput to below process discharge or flow requirements, the pump would be considered to have "failed."

The pump performance correlation is a second example that demonstrated that the project achieved the goal of providing a foundation for a first-principles prognostic algorithm.

Data Integration: This task centered on the completion of communications interfaces and the integration of the six (DSOM Operational Instrument Display, Dynamic Laser Alignment, Vibration Accelerometers, Bearing Dynamic Load Cell System, Acoustic Emission Array, and Ultrasonic Fouling Meter) independent technology information systems (TIS). This required upgrades to the SDMS Software architecture design and development of additional hardware and software interfaces. The goal was to integrate the TIS outputs into the Decision Support for Operations and Maintenance (DSOM) system to facilitate data transfer, diagnostics, and display (see Figure 6).

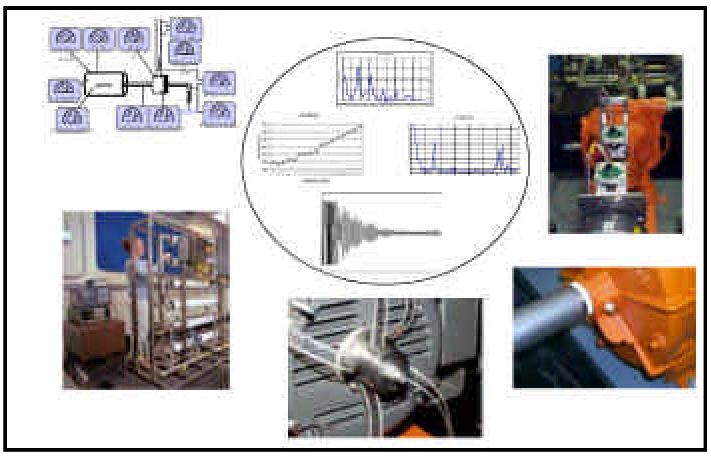


Figure 6. The graphics illustrate Multi-Technology Diagnostic Reasoning.

A software module was developed to provide a generic means to transfer spectra obtained from FFT processing from distributed processors to the DSOM system to support diagnostic analysis. Large data sets, on the order of tens of kilobytes, result from the high sample rates and the number of elemental data points required for useful frequency analysis. Transfer of such large data sets to the DSOM system would be highly inefficient particularly since the diagnostic analysis centers around only certain subsets of the data. Instead, the data sets are reduced through the distributed processing capability designed into the SDMS. The resulting data provided to the DSOM system is only that required by the diagnostic algorithms, thus reducing transmitted data by more than a factor of 100.

The DSOM diagnostic logic then selects and analyzes specific frequency bands to compare maximum peak height and integral area within the selected band with baseline data. Logic trip levels are set based on these comparisons and the result is passed to the operator via a diagnostic alert. This new integrated approach allows a prognostic evaluation to be made, and the operator can now make informed decisions based on equipment damage rates and their effect on the projected equipment residual life.

Fouling Monitor: An ultrasonic "fouling meter" was implemented on the reverse osmosis (RO) filters using a suite of 0.5 and 1.0 MHz transducers. Preliminary testing is shown in progress in Figure 3. The ultrasonic data demonstrate that pulse-echo and transmission measurements can provide a non-destructive on-line real-time monitor for filter condition assessment. Data have been shown to be sensitive to fouling layer (during both fouling and cleaning), solution concentration, membrane condition, and filter internal structure, including changes during operation. Preliminary data indicate that these can potentially be used to predict rate of change in filter condition and form the basis for the application of prognostics to the development of various forms of fouling. Similar approaches can potentially be implemented in heat exchangers and other elements in process plants where deposition or erosion is encountered.

Planned Activities

The project has met all its major goals. Data produced by the NERI SDMS experiments have been used to generate a new and definitive set of correlations linking degradation stressors to resulting degradation rates and failure prognostics. The correlations are pragmatic in their formulation and will be applicable to current as well as future generations of fossil and nuclear reactor power plants.

The methodology developed through this project points the way to real-time diagnostic/prognostic predictions that will pay significant dividends in terms of risk reduction, root cause resolution, equipment reliability, and resource management under normal and emergency conditions. The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

Concept Analysis of a Modular 50-MW(Th), Pebble-Bed, High-Temperature, Gas-Cooled Reactor for Process Heat

Primary Investigator: Dennis R. Liles, Los Alamos National Laboratory

Collaborators: Texas A&M University

Project Number: 99-188

Project Start Date: August 1999

Project End Date: September 2000

Research Objectives

The overall research objective was to develop an updated concept for a small, modular, high-temperature, helium-cooled reactor that could be used to produce hydrogen by reforming methane and steam. Such an application demonstrates the use of nuclear-generated heat as an alternative to burning fossil fuels to produce process heat.

The main goal of this study was to explore one principal option for coupling nuclear reactor heat to an endothermic chemical process: specifically, a small (50-MWth), modular, pebble-bed reactor coupled to a steam-methane reforming system. Important aspects of reactor design and safety, fuel performance, and system requirements are discussed. In particular, various nuclear fuel compositions, a change in the fuel-particle coating (from SiC to ZrC), and their influence on the reactor's physics and safety are analyzed.

Research Progress

In the high-temperature, gas-cooled reactor (HTGR) core, the use of coated particle fuel creates a double heterogeneity, both on a microscopic level (fuel particles) and on a macroscopic level (fuel element). This double heterogeneity must be considered in the neutronics calculations. Apart from a double heterogeneity, the use of helium as a coolant means that the pebble-bed HTGR core contains spaces that, because of the low density of the gas, can be considered as voids for neutronics calculations.

The major parts of detailed neutronics calculations were carried out with the SCALE and MCNP code systems. In SCALE resonance self-shielding calculations, the spatial self-shielding due to the heterogeneity is considered using Dancoff correction factors. Additional calculations were performed to determine the appropriate model for a correct description of such a complicated system as the pebble-bed core. SCALE is an extremely important tool for fuel performance evaluation because it can perform accurate fuel-depletion and fission-product-buildup calculations, thus accounting for cross-section and flux changes due to fuel irradiation. The MCNP code was used for final verification of the model.

The innovative use of composite materials such as carbon-carbon composites and zirconium as a coating for the fuel particles increases the margin of safety in high-temperature regions of the system. The proposed system is shown schematically in Figure 1; parallel components are not shown to improve clarity.

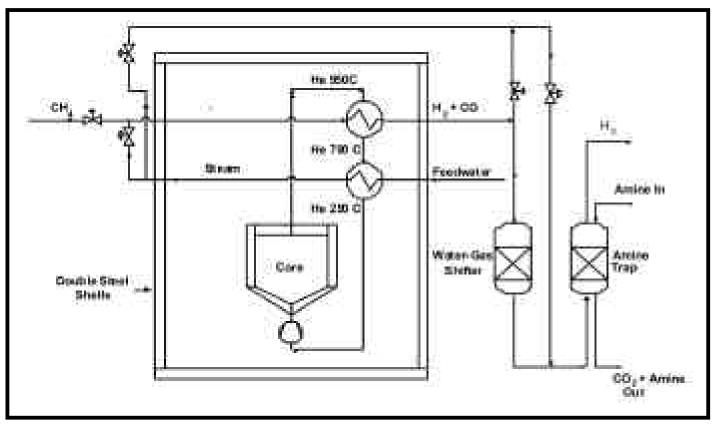


Figure 1. The schematic shows the design for a small modular HTGR for steam-methane reforming.

The primary circuit shown above consists of helium at 40 bars driven by dual circulators upward through the packed pebble bed. The helium enters at 250°C and leaves at a mixed mean temperature of 950°C. Carbon-carbon ducts connect the reactor to the first heat exchanger, the process unit. The helium enters the shell side of the heat exchanger and passes through once. Regenerative bayonet tubes contain the methane, steam, and process gases. Helium exits at 700°C and enters the steam generator, again on the shell side. U tubes contain the water-steam mixture. The helium exits at 250°C and returns to the circulators. Both the process stream and the water enter their respective heat exchangers at a pressure only a few bars over the primary system pressure to minimize stress on heat-exchanger components.

Steam-methane reforming is a catalyzed equilibrium reaction between methane (CH_4) and steam (H_2O) to produce hydrogen gas (H_2), carbon monoxide gas (CO), and carbon dioxide gas (CO_2). Additional processing can lead to either the separation of relatively pure hydrogen gas or the production of methanol. Alternatively, the product stream can be optimized to generate a mixture of carbon monoxide and hydrogen, which can be transported easily over long distances in a closed loop (so that the gas is recycled). These gases then recombine to produce methane and steam, an exothermic reaction releasing usable heat at the destination. The goal of the current project is to produce a relatively pure hydrogen gas. Steam-methane reforming is a well-established, commercial method for producing hydrogen gas in chemical processing plants. However, CO_2 is still a product of the process and must be considered. The reactor/reformer system proposed here reduces the CO_2 emission only by using a reactor to generate the process heat that eliminates burning fossil fuel. Further reduction in CO_2 emission requires application of advanced systems that have not yet been proven commercially.

Planned Activities

The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

Novel, Integrated Reactor / Power Conversion System (LMR-AMTEC Project)

Primary Investigator: Pablo R. Rubiolo, Westinghouse Electric Company LLC

Collaborators: University of New Mexico (UNM); New Mexico Institute of Mining and Technology

Project Number: 99-198

Project Start Date: August 1999

Project End Date: August 2002

Research Objectives

The overall objectives of this project are to assess the feasibility, develop engineering solutions, and determine a range of potential applications for a Novel Integrated Reactor/Energy Conversion System. The goal is to design a power supply for use by developing countries and in remote locations that is proliferation-resistant, reliable, and economical. The main features of this project are the development of a long life liquid metal reactor (LMR) (without refueling up to 10 years), and of a static conversion subsystem comprised of an Alkali Metal Thermal-to-Electric (AMTEC) topping cycle and a Thermoelectric (TE) Bottom cycle. In addition, various options of coupling the LMR with the energy conversion subsystem are being explored.

The project is being performed by the Westinghouse Electric Company LLC, which is responsible for the long-life sodium reactor development; the University of New Mexico's Institute for Space and Nuclear Power Studies, which is developing the AMTEC/TE energy conversion system; and the Institute for Engineering Research and Applications (IERA) at New Mexico Institute of Mining and Technology, which is responsible for the design of the electric converter modules and supports Westinghouse's activities related to transport safety and waste disposal.

Research Progress

The research progress and achievements are reported in this section by project participant.

Westinghouse Electric Company LLC

The work falls into three major areas:

<u>Selection of the reference LMR-AMTEC design concept</u>
Different design options were evaluated using a plant model. An Indirect Coupling (IC) plant with Alkali Metal Boilers (AMB) (see Figure 1) was chosen as the reference design, since it exhibits the best performance. The main features of the design are as follows:

- Two independent loops are employed for the IC between the LMR and the AMTEC units. No shielding is required for the secondary loop.
- Sodium and potassium are used as primary and secondary coolants, respectively.
- The net plant efficiency is 28.2 percent, when the core outlet temperature is 1,070°K.
- The LMR core is composed of 78 fuel elements (13 control rod assemblies) and 78 reflector elements. The fuel is (U,Pu)N and the cladding is made of the refractory alloy Nb-1Zr.
- The AMBs generate the potassium vapor, which is fed into the AMTEC units. The AMB can operate in once-through or in recirculation mode (with a vapor separator).
- The LMR is a pool reactor, the AMB and the primary pumps are placed inside the reactor vessel, hence excluding large LOCAs (Lost of Coolant Accident) by design.

<u>Operating parameters of the LMR-AMTEC</u>
The following plant parameters and components were determined and studied: working temperatures; flow rates and pressures; core design (fuel and cladding); alkali metal boiler design and operation; primary pumps characteristics; flow-induced vibrations in fuel elements and AMB tubes; corrosion allowance; reactor vessel design; and the in-vessel layout. A first economic anlysis of the plant was also performed.

<u>Safety features of the LMR-AMTEC</u>
The work concerning safety aspects of the LMR-AMTEC system includes the following:

- Reactivity control systems: An actively controlled absorber rod bundle assembly was adopted as a reactivity control system. Boron carbide was selected as the control rod absorber material. The studies of the reactivity behavior upon sodium removal showed an acceptable core reactivity response.
- Heat removal systems: The secondary loop, composed of the alkali metal boilers and the AMTEC units, was chosen as a normal Decay Heat Removal System (DHRS) of non-safety grade during hot and cold shutdown. In addition, a safety grade Passive Heat Removal System (PHRS) is proposed as an emergency DHRS. The PHRS of the LMR-AMTEC is activated by changes of the sodium levels that occur after the primary pumps are tripped.

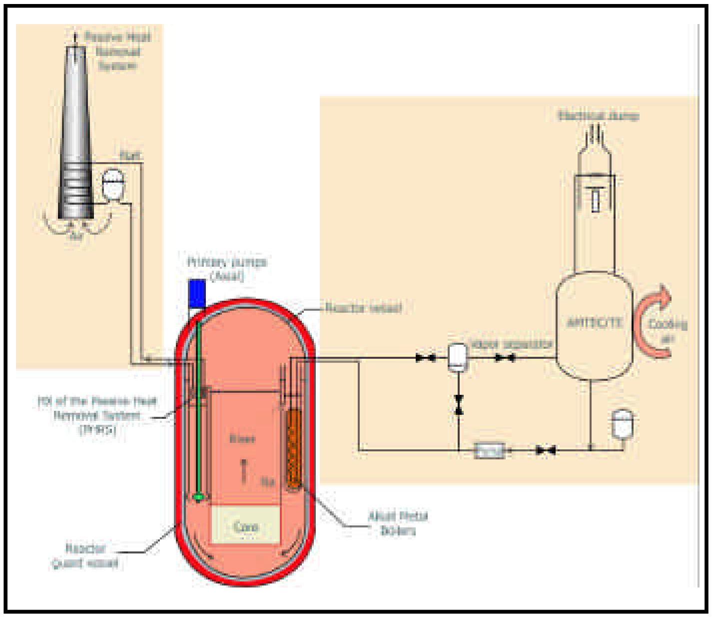


Figure 1. Schematic showing LMR-AMTEC plant design.

Institute for Space and Nuclear Power Studies (ISNPS) (University of New Mexico)

The converter, which comprises an AMTEC top cycle and a PbTe TE bottom cycle on the condenser side of the AMTEC, was designed and optimized for maximum overall conversion efficiency, for both sodium and potassium working fluids. The AMTEC topping cycle of the power conversion units delivers high power (>40 kWe) and high voltage (~400 V). The operating temperature of the evaporator (beta-alumina solid electrolyte, BASE) is approximately 1,000°K and approximately 1,121°K, for potassium and sodium working fluids, respectively. The heat rejected by the condenser of the AMTEC flows to the bottom cycle of TE modules, through a conductive coupling arrangement. The electricity generated by the TE bottom cycle, which is cooled by natural convection of ambient air, contributes between 10 percent and 30 percent of the total electric power generated by the AMTEC/TE converter units.

Performance analyses of the AMTEC/TE converter units showed that a total conversion efficiency in excess of 30 percent could be achieved, based on conservative assumptions regarding the technology of the AMTECs, and using off-the-shelf technology of lead-telluride (PbTe) TE modules. As more advances are made in the technology, higher conversion efficiencies, in excess of 34 percent for the combined AMTEC/TE converters, and a long operation lifetime of 5 to 10 years, with little degradation, would be possible in the near term.

The interfacing arrangement of the plant designs developed and investigated by the UNM-ISNPS was obtained by providing the reactor thermal power to the AMTEC/TE converter units through a heat exchanger (HX). The overall thermal and electrical performances of the plant were evaluated using a thermal-hydraulic model of the secondary loop of the LMR-AMTEC. In these designs, the secondary sodium or potassium liquid exiting the HX is partially evaporated in expanders placed in each AMTEC/TE converter. These studies showed that a Na/K plant (sodium in the primary loop and potassium in the secondary loop) could deliver a net power output of 25.4 MWe at an overall conversion efficiency of 28.6 percent, while for a Na/Na plant, the net electrical power output is 25 MWe at an overall plant efficiency of 27.7 percent. In addition, these analyses showed that the K-AMTEC/PbTe converters have an efficiency higher than that of the Na-AMTEC/PbTe converters. However, the K-AMTEC/PbTe converters deliver an electrical power output lower than the Na-AMTEC/PbTe, requiring the use of 25 percent more converter units in the Na/K plant and as a result, increasing the cost per kilowatt-hour.

Regarding the thermoelectric bottom cycle, the bottom cycle to the condenser of the top cycle was designed and thermally coupled. The use of different thermoelectric materials (instead of PbTe) for the bottom cycle was also investigated, including both single- and multisegment thermoelectric couples. Finally a performance model of the fully integrated sodium-and potassium-AMTEC/TE converters was developed. This model was used to optimize the unit's design for maximum efficiency and to investigate and determine the operation regime in which the static AMTEC/TE converters are load-following.

Additional work includes the study of different high-energy uses and the nuclear power plant options for those applications.

Institute for Engineering Research and Applications

The work performed by the IERA included topics related to transport safety, corrosion control and waste disposal of the LMR-AMTEC. Based on the selected design of the LMR-AMTEC components and the coolant types, the wastes were classified and characterized according to Code of Federal Regulation.

Planned Activities

The NERI project has been completed.

NUCLEAR ENERGY RESEARCH INITIATIVE

Direct Energy Conversion Fission Reactor

Primary Investigator: Gary Rochau, Sandia National Laboratories

Collaborators: Sandia National Laboratory; Los Alamos National Laboratory; General Atomics; University of Florida; Texas A&M University

Project Number: 99-199

Project Start Date: September 1999

Project End Date: December 2002

Research Objectives

The U.S. Department of Energy, Nuclear Energy Research Initiative (NERI) Direct Energy Conversion (DEC) project began in August of 1998 with the goal of developing a direct energy conversion process suitable for commercial development. In the first two years of the project fission processes that capture the energy of the fission fragment were examined through electromagnetic or magnetohydrodynamic principles. Roughly two-thirds of the project has been completed and investigators believe that a viable direct energy device is possible. Three concepts are under conceptual development: a Fission Electric Cell using magnetic insulation, a Magnetic Collimator using magnetic fields to direct fission fragments to collectors, and a Gas Vapor Core Reactor using magnetohydrodynamics (MHD) to generate electrical current. The concept-definition effort has focused on detailed examination of the physics of each concept and on building an engineering model to examine the design options for a power plant. The original plan was to select only one concept at the end of the second phase, but it is now believed that too many details require further exploration before such a selection can be made.

Research Progress

During the first phase of the study, nine different concepts were investigated. These concepts were analyzed and ranked to select the top three. All the selected concepts use a magnetic field in various configurations to extract energy from the fission fragments. The Quasi-Spherical Magnetically Insulated Fission Electric Cell or Fission Electric Cell (FEC), the Fission Fragment Magnetic Collimator (FFMC), and the Gaseous Vapor Core (GVC) reactor have been investigated in greater detail during Phase 2 with the objective of understanding how these concepts might be scaled to power reactor. Phase 3 efforts have focused on completing the concept definition and outlining proof of principle experiments.

Fission Electric Cell: This concept (Figure 1) originally used spherical cells with the fissioning cathode placed at the center. High-intensity, shaped magnetic fields trap the

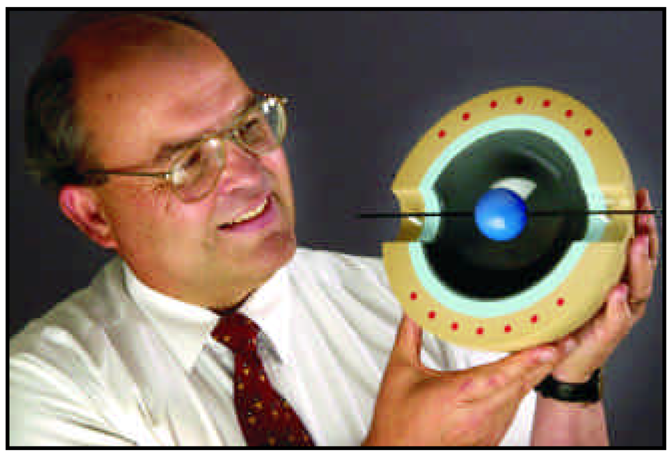


Figure 1. The individual in the photograph is holding a model of the Fission Cell under development.

electrons near the cathode allowing the more massive fission fragments to reach the anode and deposit their charge.

Because a spherical geometry maximizes the recovery of fission particles, this scheme could theoretically achieve efficiencies as high as 60 percent. Unfortunately, this design has an extremely low fuel density, making a critical assembly very difficult. Additionally, the material generating a magnetic field absorbed too many neutrons. New configurations have been developed using externally generated magnetic fields in slab, cylindrical, and spherical geometries. Overall, these devices would be very compact but have a low power density (between 0.005 and 0.015 watts per cubic centimeter). These configurations have been modeled in terms of performance, thermodynamics, and criticality.

Issues associated with this design are the size of the assembly, ability to insulate small vacuum gaps to 20 M V/cm, cathode mechanical stability, maintenance of criticality, and conversion efficiency. These issues direct researchers to look at geometries that are more easily constructed (slab and cylindrical), trading system efficiency for mechanical stability. Low efficiencies and criticality issues are guiding them to examine on-line refueling techniques to maximize fuel burn-up and utilization. Experimental designs are being studied to establish the scientific feasibility of the FEC. These experiments examine the ability to insulate small vacuum gaps, cathode stability, formation of a secondary cathode, and fission fragment charge collection. The economic feasibility of the FEC is also being studied. Initial estimates indicate that electrical power can be generated for a cost of between $0.02 and $0.09. Electromagnetic modeling of the transport of electrons in the spherical geometry is being performed during the Phase 3 study. These calculations are providing insight to the transport of electrons from cell to cell and providing a basis for experimental designs.

Fission Fragment Magnetic Collimator: This theoretical device uses magnetic fields to direct positive and negative charges to common collectors at both ends of a solenoid magnet. Fissionable material, in thin wires or threads, is placed in a parallel magnetic field inside a cylindrical vacuum chamber. As the fuel fissions, the electrons and positive fission fragments remain separated and drift to the ends of the cylinder. At the ends, the particle energy is collected in an electric-field insulated collector.

A performance model of the FFMC has been developed incorporating an electromagnetic code to track electrons and fission fragments to the collector devices. The model indicates that overall efficiency as high as 55 percent can be achieved with thin fiber-type fuel elements. This design avoids the cathode stability issue of the FEC in that high electric fields are avoided in the region of the thin fissioning fuel. Further investigation of the charge collectors is required to establish the engineering design and to identify any stability issues in the collector area. Future experiments are being defined to focus on the degree of charge separation and possible space charge effects that may limit the power density of the system.

Gaseous Vapor Core Reactor: The third approach to direct conversion uses MHD to generate electricity coupled with a more conventional steam cycle to achieve high conversion efficiency. This direct scheme uses a high-temperature gaseous core reactor to generate ionized fissioning plasma that passes through an MHD channel to generate electricity. The unprocessed heat from the MHD cycle is transferred to a superheated Brayton cycle (gas turbine) and/or Rankine power cycle (steam turbine) to achieve combined efficiencies on the order of 60 to 70 percent. The concept has several advantages over existing reactor designs in that it embodies an extremely simple reactor core design with a fully integrated fuel cycle design that includes minor actinide burning. The design has a low fuel inventory: three orders of magnitude lower than conventional light water reactors. The design has disadvantages in that material temperatures are greater than 2,500°K and more research is needed in the fission-enhanced conductivity of the vapor fuel (crucial to the MHD efficiency) in addition to the immature state of MHD power conversion.

Future work will focus on material issues, conductivity of vapor fuel, and design of the purification system.

Planned Activities

The final phase of the project is now being completed. Lacking sufficient resources for performing critical technology experiments, the team cannot make an effective selection of one concept on the basis of physical data. Consequently, the first activity in the third and final year of the project is to move toward a "Final Concept Definition" for each of the three concepts suitable for refinement and publication.

Equally important to the final project phase is the definition of experiments to provide preliminary proof of principle experiments for each of the concepts. These experiments will highlight the path forward for developing the concepts into a viable energy source. Throughout the entire process, the intent of the project has been to provide the technical basis for an alternative commercial energy source. The final report is expected to provide continuity between this effort and any future design development and commercialization efforts.

NUCLEAR ENERGY RESEARCH INITIATIVE

Novel Investigation of Iron Cross Sections via Spherical Shell Transmission Measurements and Particle Transport Calculations for Material Embrittlement Studies

Primary Investigator: Steven M. Grimes, Ohio University

Collaborators: University of Florida; National Institute of Standards and Technology

Project Number: 99-228

Project Start Date: August 1999

Project End Date: December 2002

Research Objectives

The principal objective of this project is to study neutron scattering interactions in iron as a means of investigating the well-known deficiency that exists in reactor pressure vessel neutron fluence determinations. The spherical-shell transmission method is being used, employing iron shells with different thicknesses and neutron time-of-flight measurements of the scattered neutrons, to determine precisely specific energy regions over which deficiencies in the non-elastic scattering cross section for neutron scattering in iron appear to exist. Analysis of the experimental data involves correlating the data with theoretical calculations of neutron transport through the iron spheres in order to evaluate the degree to which the calculated neutron spectra predict the measured spectra relative to different types of particle interactions. In the process, new methodologies were developed for performing neutron transport calculations that are useful to a range of transport problems.

Research Progress

Spherical shell transmission measurements were performed using neutron time-of-flight spectroscopy and accelerator-based neutron sources at selected energies greater than 1 MeV using both the $^{15}N(p,n)$ and D(d,n) sources. These measurements were conducted on two different high-purity iron spheres with shell thicknesses of approximately 3 cm and 8 cm. These two selections for sphere wall thickness were chosen in order to enhance the experiment's sensitivity to the iron cross sections, based on Monte Carlo simulations of the experiment that indicate a range of iron shell thicknesses from approximately 4 cm to 8 cm should be optimal. These measurements reveal information on the total non-elastic cross section, and various components of the non-elastic cross section for which there are neutrons in the exit channel, and provide a way of determining the quality of evaluated microscopic cross section data by an application to a macroscopic system through which neutron transport can be determined.

The larger most recently fabricated iron sphere was fabricated from high-purity iron with a low carbon content that was hot-rolled and forged to produce cylindrical billets. The iron sphere was constructed in the form of two hemispherical sections that were cut from the cylindrical billets; this fabrication work was performed using a numerical controlled lathe at Ohio University. The new sphere has some major advantages over the first sphere that was used. The new sphere was fabricated with less interior void space, which allowed the sphere to have a greater annular thickness without incurring a large increase in its outer diameter. This aspect is important relative to the alignment of beamline components and the selection of appropriate neutron beam collimation. The purity of the iron is also higher, thereby allowing for more accurate modeling of the experimental results. In addition, a new gas cell was designed to provide good charged-particle beam collimation and current integration capacity. Three concentric tubes of stainless steel constitute the major pieces. The outer tube contains a 1-mm thick gold beam stop at its end. The middle tube holds a tungsten foil approximately 5 micrometers thick at its end. The inner tube holds the final beam collimator and charge suppression. Each tube is soft-soldered to a brass cylinder that has o-ring seals. Between the inner tube's brass cylinder and the outer and middle tubes, there is a piece of Teflon to insulate the outer two tubes for charge collection.

Experimental runs were first started by collecting necessary calibration spectra to determine the efficiency of the NE-213 being employed. For this task, the Al(d,n) reaction was used, which was determined to be slightly better known than the B(d,n) reaction. A stopping aluminum target was used along with a deuteron energy of 7.44 MeV at an angle of 120 degrees for these calibrations. This energy and angle were those used for the source spectrum determination, which had been carried out previously relative to the ^{235}U(n,f) standard neutron cross section. Measurements of the bare source were made for the ^{15}N (p,n) reaction at angles of 0, 15, 45, 60, 90, 100, 120, and 135 degrees using the NE-213 detector. These measurements were made using 5.1 MeV protons emerging from the tungsten foil into the 3-cm gas cell (which was maintained at 1.5 atmospheres). A large number of angles was used for both the ^{15}N(p,n) and the D(d,n) source reaction work in order to provide detailed source information for the computer simulations. Additional measurements were also made with the small sphere and the large sphere for this source at angles of 0, 45, 90, 120, and 135 degrees using a NE-213 detector. A series of additional source measurements were conducted for the D(d,n) reaction using the NE-213 detector. These measurements were made at angles of 0, 15, 45, 60, 90, 100, 120, and 135 degrees for 3.0 MeV, 5.0 MeV, and 7.0 MeV deuterons emerging from the tungsten foil into the gas cell, which was maintained at 2 atmospheres. Corresponding runs were also made at zero degrees with no gas in the cell. Runs were then made at these three deuteron energies with the large sphere surrounding the source. For each deuteron energy, measurements were made at 0, 45, 90, 120, and 135 degrees. Random and differential linearity runs were performed to calibrate the time per channel.

Detailed particle transport calculations have been performed to optimize the experiment, to improve the accuracy of the experimental data, and to permit testing of neutron cross sections for comparing the measurements to calculations of the neutron transport through the shells. A series of time-dependent neutron transport calculations were first employed to investigate different experimental configurations in order to optimize the experiment. For this task, the A^3-MCNP (Automated Adjoint Accelerated MCNP) computer code and the three dimensional Parallel Environment Neutral-particle TRANsport (PENTRAN) code were utilized in a parallel computing environment. For this work, two PC clusters (PCPEN and PCA3MC) are in use at the University of Florida. The experimental data is being analyzed using both Monte Carlo and deterministic discrete ordinates neutron transport techniques in order to obtain information about energy regions where problems may exist with accepted iron cross section evaluations.

Additional work on modeling the neutron transport has focused on improving the accuracy of the model and speeding up the calculations. The Monte Carlo model has been revised to accommodate a more accurate representation of the geometry and materials existing in the source region and the collimator. A simplified treatment of the source gas cell has been applied, and the source has also been modified to allow for full rotation, which simulates the accelerator's beam swinger facility. The Monte Carlo model has been further updated to better incorporate the experimentally determined neutron detection efficiency, which is used to scale the scoring of the detector tally. The experimental runs are being analyzed using the revised geometry and treatment of the detector efficiency. Since the Monte Carlo simulation times for these models are excessively long, effort has been made to identify that part of the simulation that is the most time consuming. It was determined that although approximately 80 percent of the simulation time is spent tracking neutrons throughout the iron sphere due to the large number of collisions in this region, a large number of these collisions result in neutron tracks that fail to scatter into the detector. More specifically, only about 1 out of 100,000 collisions in the sphere are associated with "neutrons" that are subsequently detected. This result has then been the focus of further study resulting in the development of a suitable Monte Carlo variance reduction methodology, and it was concluded that the most effective variance reduction methodology for this problem is source biasing. In this method, the source neutrons that travel towards the detector are given more weight while the neutrons traveling backwards are weighted much less. This approach results in greater sampling of those neutrons that are more likely to contribute to the detector response. Because of parallel processing and the implementation of variance reduction techniques, a speed-up of almost two orders of magnitude has been achieved while obtaining a statistical uncertainty of about 5 percent. Preliminary results show good agreement between the experimental data and the calculated flight times for those cases where the iron sphere is not in place, thereby indicating that the model of the neutron source and time-of-flight environment are adequate. This is shown, for example, in Figure 1 for the D(d,n) reaction with 5 MeV deuterons and an angle of zero degrees.

Planned Activities

On-going and planned activities are focused on analyzing and understanding the results obtained from those cases in which the iron sphere was in place surrounding the neutron source. In Figure 2, the same conditions are in place as for Figure 1, but with the smaller sphere in place. One comparison is shown in which the simulated spectrum follows the same trends in shape as the experimental data. However, it is clear from

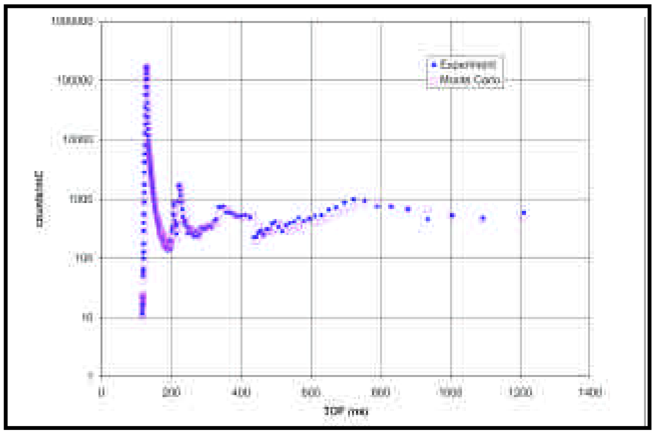


Figure 1. The graph indicates there is close agreement between the experimental and theoretical spectra when the iron sphere is not in place near the neutron source.

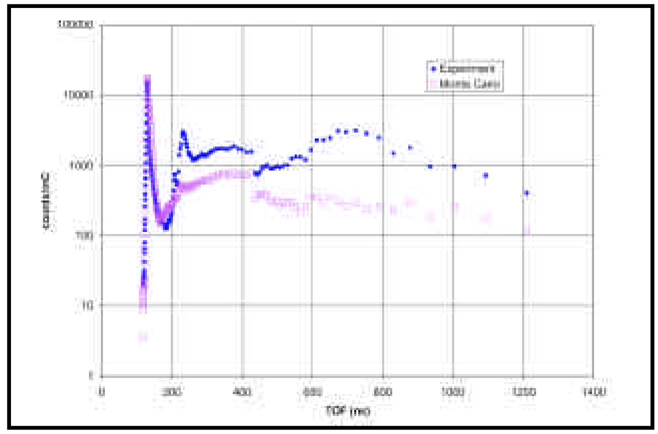


Figure 2. When the small iron sphere is in place surrounding the neutron source, there are significant differences between the experimental and theoretical spectra.

the figure that significant differences exist between the two (calculated and experimental) spectra. The process is underway to attempt to identify the source of these differences, and to determine whether they represent artifacts of the simulation process, or are indeed a manifestation of the result of deficiencies with the evaluation of the cross sections for iron.

NUCLEAR ENERGY RESEARCH INITIATIVE

High-Efficiency Generation of Hydrogen Fuels Using Nuclear Power

Primary Investigator: Lloyd C. Brown, General Atomics

Collaborators: University of Kentucky; Sandia National Laboratories-Albuquerque

Project Number: 99-238

Project Start Date: August 1999

Project End Date: October 2002

Research Objectives

Hydrogen is an environmentally attractive transportation fuel that has the potential to displace fossil fuels when coupled with fuel cells. Fuel cells are more efficient than conventional battery/internal combustion engine combinations and do not produce nitrogen oxides during low-temperature operation. Contemporary hydrogen production is primarily based on fossil fuels-more specifically on natural gas. When hydrogen is produced using energy derived from fossil fuels, there is little or no environmental benefit. Currently, there is no large-scale, cost-effective, environmentally attractive hydrogen production process available for commercialization.

The objective of this research is to find an economically feasible process for the production of hydrogen, by nuclear means, using an advanced high-temperature nuclear reactor as the primary energy source. Hydrogen production by thermochemical water splitting, a chemical process that accomplishes the decomposition of water into hydrogen and oxygen using only heat or, in the case of a hybrid thermochemical process, by a combination of heat and electrolysis, could meet these goals.

Research Progress

Phase 1 of the project concentrated on finding a thermochemical cycle suitable for development into an economic process for the production of hydrogen based on thermal energy from an advanced nuclear reactor. An exhaustive literature search was performed to locate all thermochemical water-splitting cycles. Thermochemical water-splitting is the conversion of water into hydrogen and oxygen by a series of thermally driven chemical reactions. The 115 cycles located were screened using objective criteria, to determine which can benefit, in terms of efficiency and cost, from the high-temperature capabilities of advanced nuclear reactors. The 25 cycles remaining after the preliminary screening were subjected to more rigorous analysis. As part of this second stage screening process, detailed investigations were made into the viability of each cycle. The most recent papers were obtained for each cycle, thermodynamic calculations were made over a wide temperature range, and each chemical species was considered in each of its potential forms (gas, liquid, solid, and aqueous solution). As a result of this analysis, two cycles were rated far above the others: Adiabatic UT-3 and sulfur-iodine cycles. The sulfur-iodine (S-I) cycle was selected for detailed evaluation after considering the advantages and disadvantages of each cycle. Although the S-I cycle is projected to have a significantly higher efficiency than the Adiabatic UT-3 cycle, there were other reasons for selecting the S-I cycle for this work:

(1) An extensive analysis of the Adiabatic UT-3 cycle was recently completed in Japan, whereas the last complete flowsheet for the S-I cycle was developed in 1981.

(2) Major process improvements have been suggested in the literature for the S-I cycle, which have not been incorporated into the flowsheets.

(3) The UT-3 cycle requires major materials development if it is to operate in the proposed mode.

A very simplified schematic flow diagram of the sulfur-iodine cycle is shown in Figure 1.

The diagram indicates standard state thermodynamics of the chemical reactions at the indicated temperatures but much of the work on the process involves the non-ideality of the real thermodynamics. In fact, the key to the process is that the extreme non-ideality of the system separates the hydrogen iodide from the sulfuric acid in the presence of an excess of iodine.

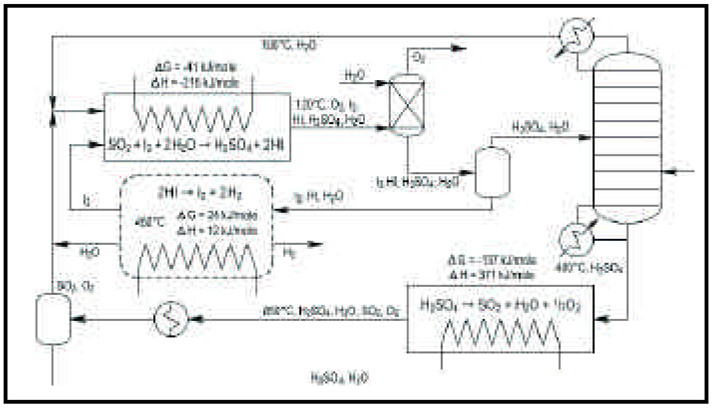


Figure 1. The schematic is a simple flow diagram of the sulfur-iodine thermochemical cycle for potentially economical production of hydrogen for transportation fuel cells.

The major undertaking during the last two years of the project was the definition of a chemical process, based on the S-I cycle, and coupled to an advanced nuclear reactor. The end result will be a flowsheet for the process, an equipment list with sizing information, and a preliminary cost estimate for the production of hydrogen from the reactor/chemical plant complex.

The last complete flowsheet for the S-I cycle was developed in 1981. This was in the early days of chemical process simulation and neither the computer algorithms nor the chemical thermodynamic models available at that time were capable of describing the very non-ideal chemistry of the cycle. Although attempts were made to use computer simulation to describe the sulfuric acid portion of the process, the flowsheet was ultimately developed using hand calculations based primarily on individual experimental data points.

Even now, after the development of chemical process simulation codes, convergence is quite difficult for very non-ideal chemistry. The lack of suitable thermodynamic models also remains a major impediment to process development. Non-ideal thermodynamic models are now being developed for common systems, but it was necessary for researchers to develop a thermodynamic model for sulfuric acid since the range of operating conditions encountered in the S-I process is far different from that encountered in normal chemical processing. Most sulfuric acid processing is performed at low temperatures. Thermodynamic models were available for sulfuric acid at low concentrations at moderate temperatures, and at high concentrations at low temperatures. However, no model was available at the high concentrations and temperatures of the S-I process.

An electrolytic non-random, two-liquid (ELECNRTL) model of sulfuric acid was developed by regressing the available thermodynamic data. Much of the pertinent data was not widely available and had not been used in previous attempts to model sulfuric acid. Aspen Plus was then used to model the sulfuric acid concentration and decomposition steps of the process. An upper bound on the hydrogen production efficiency of 61 percent was obtained by assuming that no heat would be required for the other process steps. Any heat required for the HI processing and hydrogen production step will decrease the efficiency of this value.

The system $HI/I_2/H_2O$ is even more problematic. Data was readily available for the pure components and for the two-component system, HI/H_2O, but there was a paucity of other data for the system. However, the mutual solubilities of I_2 and H_2O were available along with a few liquid-liquid equilibria measurement for the high HI two phase region and some gross vapor pressure measurements of three component mixtures. The gross vapor pressure measurements were compromised by the equilibrium decomposition of HI into H_2 and I_2. Nevertheless, it was possible to regress the data to obtain an ELECNRTL model for the system. Figure 2 shows a phase diagram for the system $HI/I_2/H_2O$ calculated from this model.

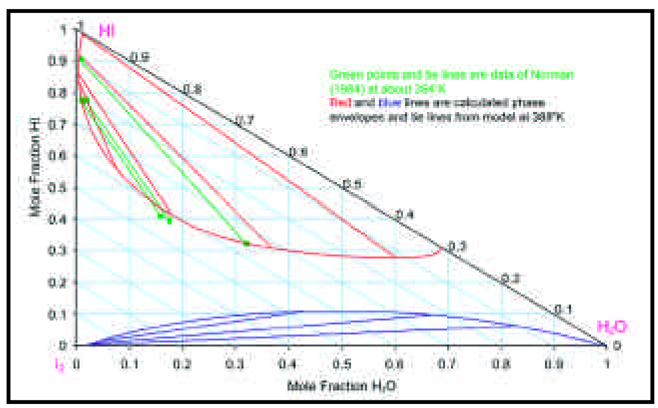


Figure 2. The figure illustrates a HI-I_2-H_2O ternary diagram at about 390°K.

Two alternative versions of the $HI/I_2/H_2O$ portions of the flowsheet are being investigated. The first is a modification of the process proposed in the original flowsheets, which avoided the complexity of the $HI/I_2/H_2O$ equilibrium by extracting water away from the HI/I_2 with concentrated phosphoric acid. This process step was efficient but very capital-intensive. Vapor recompression was used to separate the water from the phosphoric acid. The use of co-generation of electricity and hydrogen production was investigated for increasing the overall efficiency of the combined operation while decreasing the capital cost. Electricity can be produced at 51 percent efficiency using a Brayton cycle and 850°C helium from a high-temperature nuclear reactor nuclear. If it was assumed that that the waste heat from the Brayton cycle was used to separate the water from the phosphoric acid, the overall co-generation process efficiency reached 52 percent. Unfortunately, under this scenario, most of the energy goes into electricity production and only 2 percent of the total plant output is hydrogen. This is a very acceptable efficiency for a niche market but the ratio of hydrogen to electricity is too low to support a hydrogen economy. The ratio of hydrogen to electricity can be increased at the expense of the overall efficiency. At an overall efficiency of 50 percent, hydrogen represents 7 percent of the energy output and at 47 percent efficiency—the efficiency of the original hydrogen process—hydrogen is 14 percent of the output. At this level, a significant fraction of the fuels for surface transportation could be provided by hydrogen from nuclear reactors, if all of the Nation's electricity was produced from the same reactors.

The second $HI/I_2/H_2O$ process option employs reactive distillation of the three-component mixture to produce the hydrogen. This process option should have a significantly reduced capital cost compared with the other option but it has not yet been possible to converge the model.

Meanwhile, procedures for estimating the capital cost of the process plant have been adapted to the materials and unit operations of the S-I process. The procedures have been exercised with an early version of the flowsheet and are ready for use when convergence of the latest version of the flowsheet is completed.

Sandia's nuclear engineers evaluated alternative nuclear reactor concepts to select a reactor concept to be matched to the S-I process. Beginning with a list of nine reactor types, the most promising configuration of each was selected and evaluated with respect to its potential for powering the thermochemical cycle. The results are provided in Table 1.

It was determined that the reactor heat source should not in itself present any significant issues related to design, safety, operation, or economics. Pressurized and boiling water reactors, organic-cooled reactors, and gas-core reactors were found to be unsuitable for the intended application. Although alkali metal-cooled and liquid-core reactors are potential candidates, they present a significant development risk for the intended application. Heavy metal-cooled reactors and molten salt-cooled reactors show promise in being capable of meeting the requirements. However, the cost and time required for their development are uncertain and may be appreciable. Gas-cooled reactors have been successfully operated in the required 900°C coolant temperature range, and do not present any obvious design, safety, operational, or economic issues. The study concluded that a Gas-cooled reactor, employing helium as the coolant, be matched to the S-I process.

Table 1. Evaluation of Nine Reactor Types for Use with the S-I Process.

Reactor Type	Recommendation
1. Pressurized Water Reactors	Unsuitable (Insufficient temperature)
2. Boiling Water Reactors	Unsuitable (Insufficient temperature)
3. Organic-Cooled Reactors	Unsuitable (Insufficient temperature)
4. Alkali Liquid Metal-Cooled Reactors (lithium-cooled)	Potential
5. Heavy Liquid Metal-Cooled Reactors (lead-bismuth cooled)	Promising
6. Gas-Cooled Reactors (helium cooled)	Recommended
7. Molten Salt-Cooled Reactors ($2LiF$-BeF_2 cooled)	Promising
8. Liquid-Core Reactors (Molten Salt-Core)	Potential
9. Gas-Core Reactors	Unsuitable (Unacceptable development risk)

Planned Activities

The third phase of the project is almost complete. The remaining scheduled tasks are the completion of the $HI/I_2/H_2O$ portion of the flowsheet, and the generation of a cost estimate for the production of hydrogen using nuclear energy.

NUCLEAR ENERGY RESEARCH INITIATIVE

Smart Equipment and Systems to Improve Reliability and Safety in Future Nuclear Power Plant Operations (Smart-NPP)

Primary Investigator: Felicia A. Durán, Sandia National Laboratories

Collaborators: Framatome ANP; Korea Power Engineering Company; Pennsylvania State University; Massachusetts Institute of Technology; Westinghouse Nuclear Automation

Project Number: 99-306

Project Start Date: August 1999

Project End Date: December 2002

Research Objective

The goal of this research is to design, develop, and evaluate an integrated set of tools and methodologies that can improve the reliability and safety of advanced nuclear power plants (NPPs) through the introduction of smart equipment and predictive maintenance technology. This will ultimately aide in the reduction of construction, maintenance, and operational costs.

To accomplish the goal, the following activities were carried out under the Smart Equipment project:

- Identified and prioritized NPP equipment that would most likely benefit from adding smart features.
- Developed a methodology for systematically monitoring the health of individual pieces of equipment implemented with smart features (i.e., "smart" equipment).
- Developed a methodology to provide plant operators with real-time information through smart equipment Human-Machine Interface (HMI) to support their decision making.
- Demonstrated the methodology on a selected component.
- Expanded the concept to system and plant levels that allow communication and integration of data among smart equipment.

For this project, smart equipment embodies elemental components (e.g., sensor, data transmission devices, computer hardware and software, HMI devices) that continuously monitor the state of health of the equipment in terms of failure modes and remaining useful life, in order to predict degradation and potential failure and inform end-users of the need for maintenance or system-level operational adjustments.

Research Progress

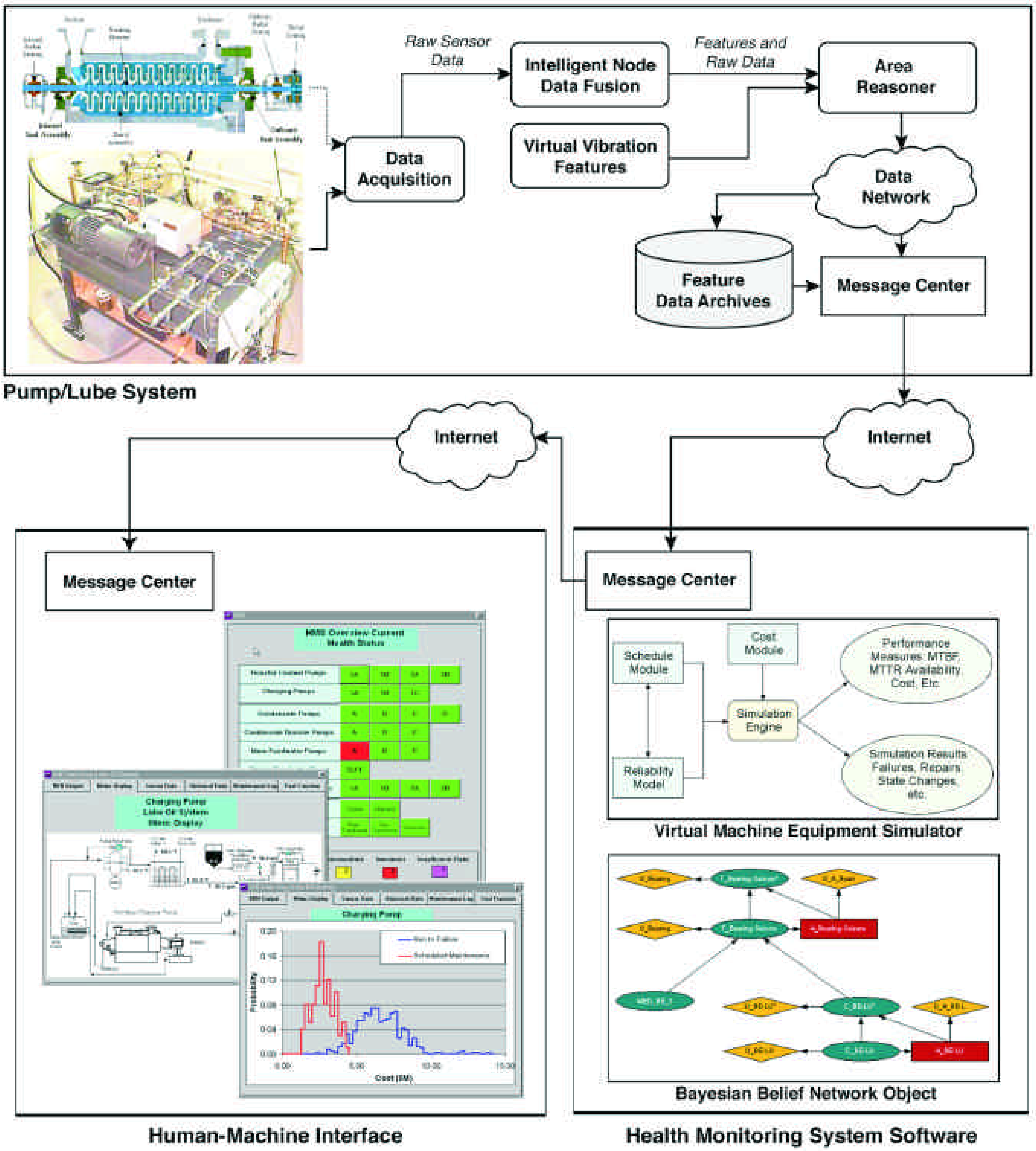
During the course of this project, the Smart-NPP team has accomplished the following:

- Completed a system prioritization study for smart equipment applications, prioritized both pressurized water reactor (PWR) and boiling water reactor (BWR) systems, and selected a high-energy, horizontal centrifugal pump as a demonstration component for a Health Monitoring System (HMS).
- Developed an HMS architecture, using Bayesian Belief Networks (BBNs) and completed detailed BBNs to relate sensor features to fault conditions for the charging pump and its lube system.
- Completed a Beta Version of the Virtual Machine Equipment Simulator (VMES) to provide a capability to simulate equipment behavior over future time intervals in order to evaluate the overall benefits to system performance from designing in smart features as well as the consequences of alternative maintenance options. Data collected on PWR equipment performance from 1990-1995 was analyzed to test the VMES and to illustrate its capability for simulating equipment failures and maintenance. The purpose of this example was to test the reliability simulation capability of VMES and to provide a limited validation of its simulation algorithms. Additionally, a sample scenario was evaluated to illustrate the use of VMES in evaluating alternative scenarios that could be considered in response to an HMS notification of a

pending equipment failure. The basis for selecting the preferred scenario will be the cost of electricity not generated as a result of a scheduled or forced outage.

- Reviewed and assessed sensor technology and instrumented the lube system test bed with sensors, including a PC104 smart sensor, that provide real-world data over the Internet to the HMS.
- Reviewed and assessed HMI technology and then developed and demonstrated an HMI as part of the integrated system.
- Instrumented the test-bed with sensors, including a PC104 smart sensor, and will provide real-world data over the Internet to the HMS.

Figure 1. Integrated Smart Equipment Demonstration Health Monitoring System Architecture



- Implemented the architecture and messaging format for the HMS to accommodate remote communication and integration among the elements of the systems.
- Established a collaborative agreement with the Korea Power Electric Company (KOPEC) for a parallel KOPEC project to develop an HMS for a control rod drive mechanism (CRDM).
- Procured the use of a demonstration test-bed of the pump lube system built at Pennsylvania State University, as the physical real world system.

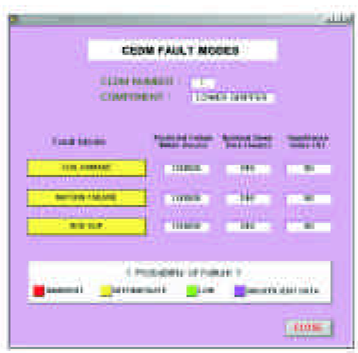
In the final phase of the project, the Smart-NPP team completed project activities to achieve the project goals. Final demonstration versions of each element of the HMS were completed, demonstration scenarios were developed to exercise the system, and the Smart-NPP team presented an integrated demonstration HMS (Figure 1) for a high-energy, horizontal, centrifugal charging pump. The demonstration HMS can replicate faults for a physical, real-world system as well as implement other faults "virtually"; process sensor features and relate these to system fault conditions; generate consequences of alternative maintenance options; and provide indications and details of system health and maintenance options on an HMI.

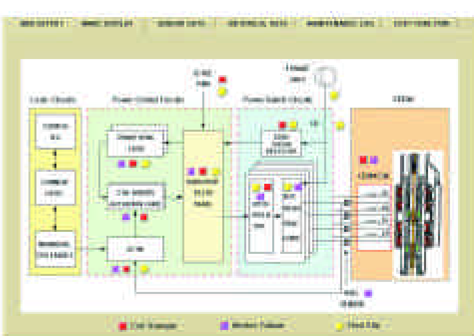
Additionally, collaboration with KOPEC continued, and the KOPEC team completed a successful application of the Smart-NPP methodologies to develop a demonstration HMS for the CRDM (Figure 2).

Planned Activities

The NERI project has been completed.

Figure 2. KOPEC Smart Equipment System for a Rod Control System.





NUCLEAR ENERGY RESEARCH INITIATIVE

Continuous-Wave Radar to Detect Defects within Heat Exchanger and Steam Generator Tubes

Primary Investigator: Gary E. Rochau, Sandia National Laboratories (SNL)

Collaborators: New Mexico State University (NMSU)

Project Number: 99-308

Project Start Date: September 1999

Project End Date: January 2003

Research Objectives

The goal is to design, fabricate, and demonstrate a radar system that will be translated internally through the steam generator tubing of nuclear power plants to find defects within the tube walls. The primary technical objective of this tool is the detection of incipient cracks that are about 20 percent, or less, of the tube-wall thickness. This will be about a two-fold improvement over present eddy-current technology. The second goal is to provide 100 percent volumetric examination of the tube wall at a translation speed of 40 inches/second. Thirdly, pattern-recognition algorithms will be developed to classify and size defects.

Research Progress

The research on this project is performed in four parallel tracks:

(1) Mechanical design of the probe system,

(2) Computational modeling of the electromagnetic performance of the probe,

(3) Laboratory prototyping of the probe system, and

(4) Electronic data collection.

The mechanical design of the probe and the test fixture was assigned as a problem for senior students in a capstone, mechanical engineering senior design class at NMSU. An early effort was to build a scaled-up version of the instrument and measure its performance in a large metal tube, and to then build a smaller version. The two different sized instruments and three types of material for their fabrication were examined, both structurally and electro-magnetically, and a 6-foot length of Inconel 600 was obtained with an ID of 3.438 inches and a 0.1-inch wall thickness. The test fixture has been fabricated, and a commercial probe translation system has been integrated into the fixture. The design of the scaled-up prototype probe, together with its centering system, is complete (see Figure 1). Prototype units for the transmitter power amplifier as well as the ferrite rod antenna have been designed and fabricated. On the receiver side, a monopole antenna and amplifier have been designed and a prototype unit fabricated. Data communication circuits have been designed, built, and tested to transmit command and control data to the probe and collect data from the probe via fiber optics. Mechanical systems for driving the probe have also been designed and implemented to complement a Z-tech commercial drive system.

The project was described at the Sixth Balance-of-Plant Heat Exchanger NDE Symposium in June 2000, to solicit industrial inputs. U.S. Patent No. 6,271,670, "A Radar Back-Scatter Probe to Detect External Cracks from within a Metal Tube," was issued on 7 August 2001 and assigned to SNL. In the same month, a paper on "In-Tube Radar" was presented by researchers from both SNL and NMSU to the International Conference on Structural

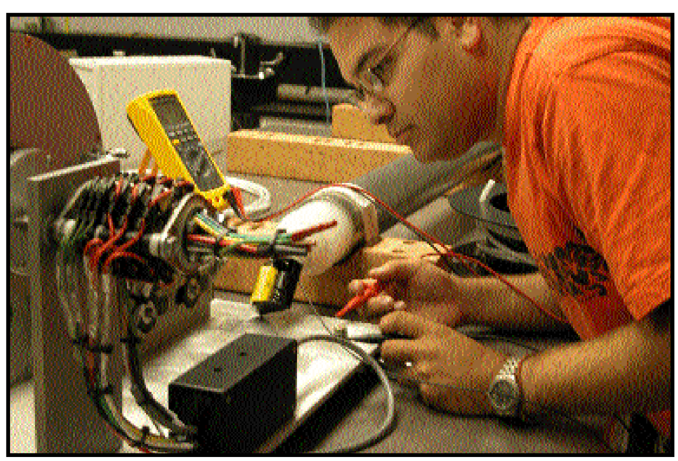


Figure 1. A scientist works on the In-tube Radar Large Scale Experiment to detect cracked tubing in nuclear power plant steam generators.

Mechanics in Reactor Technology (SmiRT 16) in Washington, D.C.

The SNL code, CTUBE, which is used to compute the internal distribution of the electric field, now provides results for unsymmetrical conditions so that it is possible to make calculations for an ITR that is tilted within a steam generator tube. CTUBE has been used to predict the cross talk voltages that will be induced upon the monopole detector. As a result, the computation of both the internal and backscattered fields that are necessary to predict Signal-to-Cross talk ratios have been done at SNL. While acceptable signals can be calculated, the intensity is extremely sensitive to alignment and requires high sensitivity equipment to detect defects in a small tube. Consequently, very specialized detector construction is required.

The laboratory design of the probe has proceeded according to schedule. The electronic and electromagnetic design of the probe, using the best technology available, continues to evolve to improve signal quality and reduce the noise observed in testing. The probe design is extremely sensitive to shielding and this has become a focus of the design revisions. The requisite design for the probe may exceed the current capability of the developers. To eliminate noise and spurious signals, an electro-magnetically sealed system with ultra-precision circuitry may be required.

The project has had significant student involvement at New Mexico State University. Four Master's degree students have been supported by the project (one in Electrical Engineering and three in Mechanical Engineering). Three senior engineering capstone design teams have worked on various aspects of the project including probe design, mock up prototype design, and electrical design, and a total of 24 undergraduate students (in Electrical Engineering, Mechanical Engineering, Industrial Engineering, and Technical Writing) have been partially supported by the grant and have made significant contributions towards the advancement of the project.

Planned Activities

A demonstration of the scaled probe, using the best available technology, is planned for December 2002.

NUCLEAR ENERGY RESEARCH INITIATIVE

Study of Cost Effective Large Advanced Pressurized Water reactors that Employ Passive Safety Features

Primary Investigator: James W. Winters, Westinghouse Electric Company LLC

Project Number: 00-023

Project Start Date: August 2000

Project End Date: September 2003

Research Objectives

On December 16, 1999, the U.S. Nuclear Regulatory Commission (NRC) issued Design Certification of the AP600 standard nuclear reactor design. This culminated an eight-year review of the AP600's design, safety analysis, and probabilistic risk assessment. The AP600 is a 600-MWe reactor that utilizes passive safety features which, once actuated, depend only on natural forces such as gravity and natural circulation to perform all required safety functions. These passive safety systems result in increased plant safety and have also significantly simplified plant systems and equipment, resulting in easier plant operation and maintenance. The AP600 meets NRC's deterministic safety criteria and probabilistic risk criteria with large margins.

The large safety margins of the AP600 can be attributed to the performance of the passive safety systems in response to accidents. An extensive AP600 test program was performed to provide confidence in the ability to adequately predict the performance characteristics of the passive safety systems as required by 10 CFR 50. This test program consisted of separate effects and integral systems tests of the passive safety systems and is well-documented in NUREG-1512, Final Safety Evaluation Report Related to Certification of the AP600 Standard Design. Westinghouse used the test programs to develop analytical computer codes that can predict with adequate certainty, the performance of the passive safety systems in response to design basis and beyond design basis accidents. In addition to the extensive test program conducted by Westinghouse, the NRC also performed confirmatory tests and analyses at both the APEX test facility at Oregon State University and the ROSA test facility at the Japan Atomic Energy Research Institute. As a result, the Westinghouse computer codes were validated as sufficient for use in performing accident analyses in accordance with the requirements of 10 CFR Part 50 and Part 52. In addition, the NRC performed independent analyses of the AP600 using different analysis codes to confirm the adequacy of the AP600 design as well as the AP600 safety analysis presented in the AP600 Standard Safety Analysis Report. These independent analyses also confirmed the large safety margins exhibited in the AP600.

Westinghouse is developing a larger version of the AP600 called the AP1000. The AP1000 design is based largely on the AP600. It employs passive systems that operate in the same manner as the AP600 passive systems. The AP1000 is being designed to meet NRC's regulatory criteria in a similar manner to that found to be acceptable for the AP600. The AP1000 is being designed to meet NRC's deterministic safety criteria and probabilistic risk criteria with large margins.

Westinghouse intends to certify the AP1000 standard plant design under the provisions of 10 CFR Part 52. To that end, Westinghouse submitted an application for Design Certification of AP1000 to the NRC on March 28, 2002. This NERI program provided support for development and analysis of some areas of the design that are included in the Design Certification application. AP1000 design features, as they relate to Design Certification, are included in the AP1000 Design Control Document (APP-GW-GL-700).

Research Progress

AP1000 uses a canned motor pump for its reactor coolant pump. Canned motor pumps are used in the U.S. Navy's nuclear program and are part of the AP600 design. The pump and motor size required for AP1000 is an extension from current practice. The plant designers worked with the pump designers to develop a pump specification that met plant requirements while minimizing the pump design extension.

AP600 is designed and certified based upon a given version of the ASME Code and other applicable national consensus standards. AP1000 should be based upon more current versions of those standards. This NERI program partially supported a study to determine which version of the ASME Code will be the basis for AP1000 and the technical basis that was provided to NRC to justify use of this version.

The steam generators for AP1000 are larger than those for AP600 and are based upon the replacement units for Arkansas Nuclear Unit 1. A unique specification was prepared to account for the AP1000 thermal hydraulic operating conditions and for the channel head mounted reactor coolant pumps.

Safeguards data packages were prepared to support safety analyses and the safety analyses themselves were performed. This NERI program helped support analysis in two areas. The results of all safety analyses are included in the AP1000 Design Control Document. The Phase 2 report for this NERI program provides examples of the nature and results of the two supported efforts for AP1000.

In 2002, the NRC prepared and transmitted 700 Requests for Additional Information (RAIs) to Westinghouse. This completes the generation of RAIs and Westinghouse will have answered them all by December 2, 2002. A large number of these RAIs relate to safety analysis. They request additional information on code development, analysis techniques, model development, assumptions, relevant equations, applicability of tests, and other details. There are no RAIs that question the basic safety of AP1000. All RAIs do require a thoughtful, documented answer for NRC to use in its development of the AP1000 Safety Evaluation Report.

In conclusion, this NERI program provided support for representative design certification activities. These activities are unique to AP1000, but are representative of research activities that must be driven to conclusion to realize successful licensing of the next generation of nuclear power plants in the United States.

Planned Activities

In Phase 3, Westinghouse will perform the safety analyses necessary to answer Requests for Additional Information (RAIs) from the NRC. Phase 3 activities will be in the areas of Loss of Coolant Accident (LOCA) analysis, non-LOCA analysis, and containment response analysis. A final report will be issued outlining the NERI supported safety-related analysis performed for AP1000.

NUCLEAR ENERGY RESEARCH INITIATIVE

Design and Layout Concepts for Compact, Factory-Produced, Transportable, Generation IV Reactor Systems

Primary Investigator: Fred Mynatt, University of Tennessee

Collaborators: Massachusetts Institute of Technology; Westinghouse Electric Company LLC; Oak Ridge National Laboratory; Newport News Shipbuilding; Institute for Physics and Power Engineering

Project Number: 00-047

Project Start Date: August 2000

Project End Date: September 2003

Research Objectives

The purpose of this research project is to develop compact Generation IV nuclear power plant design and layout concepts that maximize the benefits of factory-based fabrication and optimal packaging, transportation, and siting. The potentially small footprint of Generation IV systems offers the opportunity for maximum factory fabrication and optimal packaging for transportation and siting. Barge mounting is an option to be considered and will offer flexibility for siting including floating installation, on-shore fixed siting, and transportation to nearby inland sites. Railroad and truck transportation of system modules will also be considered in this work. The project utilizes the work of others, including both previous efforts and current Generation IV work. This includes a previously funded NERI project to develop standards and guidelines for cost-effective layout and modularization of nuclear power plants.

Research Progress

This project was funded as a grant to the University of Tennessee (UT) and subgrant to the Massachusetts Institute of Technology (MIT) with a starting date of August 15, 2000. Following the grant processing and assignment of students, work began on the first tasks—acquisition and reviews of available designs and requirements for each reactor type. Based on that work, the project team selected three reactor types. These included a helium-cooled Modular Pebble Bed Reactor (MPBR) concept being developed at MIT, the International Reactor, Innovative & Secure (IRIS) water-cooled concept being developed by a team led by Westinghouse Electric Company, and a lead-bismuth-cooled concept to be developed by UT. Work began about February 15, 2001, on computer modeling of the reactor systems and initial plant layout concepts were completed by October 15, 2001.

Development of plant layout and modularization concepts requires an understanding of both primary and secondary systems. Work to develop the MPBR at MIT included the initial concepts for both systems. The IRIS project did not have a secondary system conceptual design nor were appropriate primary and secondary system concepts available for a lead-bismuth cooled reactor. During the second phase of the project, efforts were focused to further develop the MPBR concept, and to develop a secondary system and integrated plant concept for IRIS and a lead-bismuth-cooled integrated plant concept. There was also increased interaction in the second phase among researchers on the IRIS development team for the light water reactor (LWR) concept, those working on the Oak Ridge National Laboratory MPBR concept, and several other individuals also working on lead-bismuth-cooled reactors.

LWR Concept: The objective of this work is to develop a conceptual design and layout of the balance-of-plant (BOP) for a Generation IV nuclear power plant using the 300-MWe Westinghouse IRIS as the primary system. The motivation to develop the BOP concept is to create a layout for use in modularity and manufacturing studies. In order to determine the layout, various requirements must be taken into account, for example, standard design specifications. The research focuses on two aspects: conceptual design of the BOP Power Conversion systems for the IRIS reactor, and layout of the design concept in a 3-D Computer Aided Design (CAD) package. This will provide the information needed to carry out modularity and manufacturing simulations and design specifications of the nuclear power plant.

The preliminary, conceptual design and balance of plant for a Generation IV nuclear power plant using the Westinghouse IRIS has been developed. This plant design formed the basis of a Master of Science thesis in Nuclear Engineering at the University of Tennessee (UT) in May 2002. The system output is 1000 MW thermal and approximately 360 MW electric. It has six feed water heaters and dual reheat with one high pressure and one low-pressure tandem compound turbo-generator unit. Sizes and weights of the components and associated piping are estimated. The final layout of the plant has a footprint that is 100 meters long by 40 meters wide, and weighs approximately 7400 tons. Figures containing visualizations of plant components layout and solid modeling have been developed. The main constraints for modularity are to section the plant for barge transport,

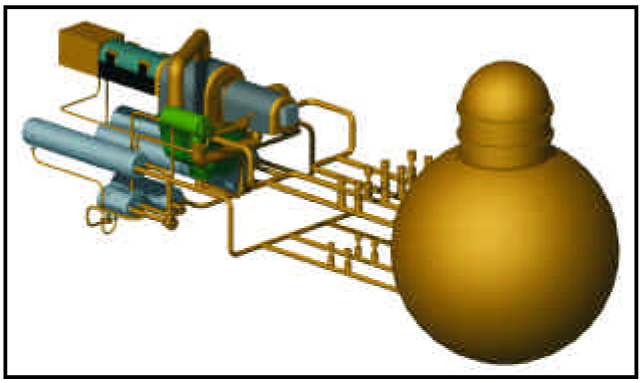


Figure 1. The figure is a visualization of IRIS containment and BOP components.

which requires that the sections must fit on a barge no larger than 400 feet long by 110 feet wide with a draft less than 9 feet. These limitations are dictated by water depth (9 feet minimum channel depth at mean low water) and specified lock sizes from the mouth of the Mississippi River northward to the Ohio River, for possible siting at Portsmouth, Ohio. The present preliminary plant design meets these criteria. Optimization of plant components to maximize efficiency and permit maintenance and repairs while minimizing capital costs has not been accomplished, although efforts to do so have been initiated for the feed water reheat system. Figure 1 shows a three-dimensional model of the IRIS containment and BOP components.

<u>MPBR Concept:</u> The MPBR concept project interfaces closely with another NERI project at MIT that involves designing the major components, including the intermediate heat exchangers, turbines, compressors, recuperators, precoolers, and intercoolers. The two teams are working together to establish layout options given the actual conceptual designs being developed. The MPBR design power level is approximately 100 MWe.

The modularity and packaging studies performed for the MPBR BOP can be broken down into several tasks:

1. System layout and design (physical layout and packaging of the plant components).
2. System concept design for increased modularity and decreased cost.
3. Advanced component design concepts for future implementation.

The first task involves defining the physical layout of the power plant itself and any transportation issues involved in its construction. The second task is concerned with making high-level trade studies of the actual system, such as the number of intercoolers, limiting temperatures, and other system parameters. The third task involves searching for advanced component concepts that may help with the other two tasks by making individual components simpler, cheaper, or more fault-tolerant.

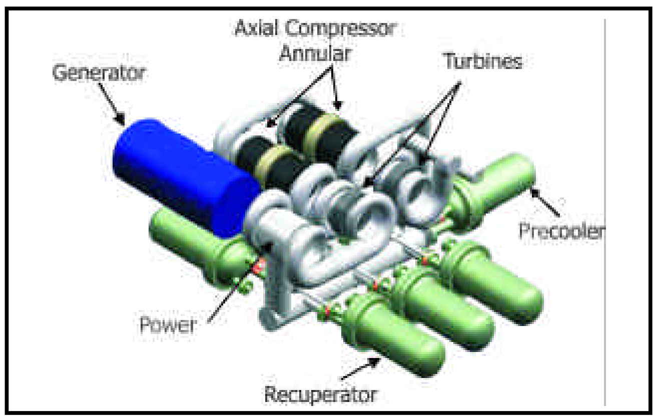


Figure 2: The diagram is a three-dimensional model of the MPBR balance-of-plant.

The proposed modular BOP system uses components and component carriers sized to fit within the limitations of truck transport. These component carriers are steel space-frames that encapsulate each component. Using this method, all the BOP components can be built in a factory and easily assembled on-site without the need for significant cranes, assembly jigs, alignment, and other special tools. Using the integral matrix structure of steel space-frames, all the necessary access hardware (e.g., catwalks, valving, flanges) can be built in the factory, further minimizing on-site assembly. Figure 2 shows a three-dimensional model of the MPBR BOP.

The second year of work focused on the development of a modular approach to the design and construction of a pebble bed reactor at the Massachusetts Institute of Technology. The basic design of the plant remains the same. New modularity features have been identified in the area of refueling systems and in integrating the basic concept with a construction plan. More detailed understanding of the component designs and sizes has affected the layout proposed. Further refinements will undoubtedly become necessary as stress calculations on piping systems and auxiliary systems are considered (instrumentation, monitoring, and control). At this point, the modularity concepts proposed still appear to be very practical and possible.

Included are several new areas of investigation regarding the modularity concept being developed. Consideration has been given to shipment of the reactor vessel, the modularity design of the online refueling system, spent fuel storage tanks, inventory control system and costs of the components of the plant including the intermediate heat exchangers, recuperators, turbines, generator, and precooler. In addition, a construction deployment plan has been developed with a vision for the actual building and for manufacturing the components of the power plant in a virtual factory with a "just-in- time" delivery system to site.

<u>Lead-Bismuth (PbBi) Concept:</u> Liquid metal breeder reactors hold particular promise for future energy supply since they offer an essentially infinite-time solution to energy production through effective utilization of fertile and fissile materials. They also can be used to recycle nearly all of the radioactive waste produced by current nuclear reactors, subsequently using the waste for energy production. Many breeder reactors have been designed and a few have been built and operated. However, most designs have an inherent problem with positive coolant voiding reactivity coefficients, and may present more risk than many scientists would prefer to accept. Results from calculations performed indicate that proper choices of thorium, plutonium, and uranium fuels, along with some changes in geometry, permit a PbBi cooled reactor to operate with a negative PbBi voiding reactivity coefficient, so that a reactor with considerably more inherent safety than previous designs can be designed and operated.

One significant advantage of PbBi as a coolant is that the reactor spectrum is relatively hard, which permits

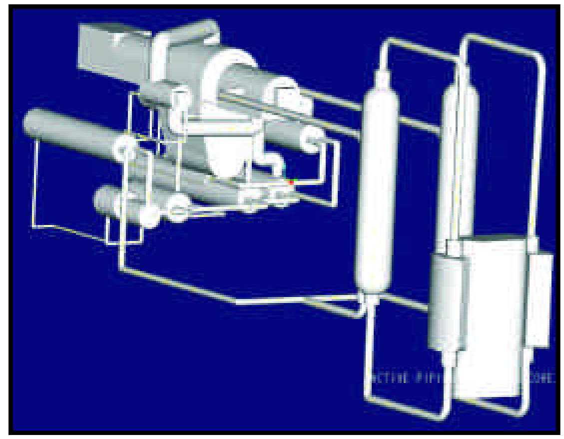


Figure 3. The figure is a three-dimensional rendition of the lead-bismuth concept.

significant quantities of actinides to be used as fuel and eliminates the need to dispose of them as waste. The nuclear characteristics of this design also permit operation for at least five years without refueling, or reshuffling, since the conversion ratio can be maintained very near unity. The time between refueling is limited by performance of fuel materials rather than by the ability to sustain the chain reaction. Proliferation resistance is improved relative to the reactors in current commercial use since the Pu-239 inventory can be held constant or be diminished, depending on fuel management choices. Figure 3 shows a three-dimensional drawing of the PbBi concept.

The reactor components in the proposed design are limited in size to a range that will allow the reactor vessel be transported on a standard rail car. This limits the height and width to about 12 feet, the length to about 80 feet, and the weight to about 80 tons. This should be adequate for producing 300 to 400 MW of electricity, but will depend on optimization of primary and secondary system performance, and must satisfy all licensing requirements.

Concept development and plant layout studies of a PbBi cooled reactor are completed. It was determined that a PbBi cooled fast reactor that produces 310 MWe can be designed with components that are all rail-transportable. It was further determined that a practical PbBi cooled reactor that uses only Pu as fuel, and that has a negative voiding coefficient, probably cannot be designed without the use of leakage-enhanced fuel assemblies. However, results to date indicate that a relatively high leakage slab core that uses a combination of Pu, U, and Th for fuel does have a negative coolant voiding coefficient. The reference system design uses steam generators coupled to a secondary system designed for IRIS as part of this NERI project and has an overall efficiency of about 35 percent, which could probably be increased to about 40 percent with additional design effort. A PbBi cooled fast reactor provides a long-term option for sustainable nuclear power, and it can be operated to produce very little transuranic waste.

The work on the three plant concepts and layouts has been completed. This includes refinement of the computational models and the concept layouts. The review of all three reactor concepts by Northrop Grumman Newport News (formerly Newport News Shipbuilding) has been completed. A review of the LWR concept by Westinghouse was not performed in the second phase. The DOE has approved a no-cost extension of the Westinghouse subcontract to perform related work in the third phase.

Planned Activities

The primary effort in the third phase of this project, simulation and analysis of modular reactor fabrication and manufacturing, began on August 15, 2002. This work will be performed by an Industrial Engineering student and professor at UT with assistance and input by the professors who led the reactor concept developments. The focus is on evaluation of economies of factory fabrication versus economies of scale of site-constructed large plants. A related effort is to evaluate whether the modular approach really works in that it is feasible and cost-effective to build modules in the factory and assemble them at the site.

NUCLEAR ENERGY RESEARCH INITIATIVE

The Secure Transportable Autonomous Reactor for Hydrogen Production

Primary Investigator: David Wade, Argonne National Laboratory

Collaborators: Texas A&M University

Project Number: 00-060

Project Start Date: October 2000

Project End Date: September 2003

Research Objectives

The Secure Transportable Autonomous Reactor for Hydrogen production is a modular fast reactor intended for the mid 21st century energy market. In this system, electricity and hydrogen are employed as complementary energy carriers, and nuclear energy contributes to a sustainable energy supply based on full transuranic recycle in a passively safe, environmentally friendly and proliferation-resistant manner suitable for widespread worldwide deployment.

This is a report on progress made in the first 24 months of a three-year NERI grant to develop this concept.

Research Progress

During the first year, the basic design selections were made and neutronics and thermohydraulics feasibility were established. The reactor is a Pb-cooled, fast spectrum, TRU-Nitride-fueled, 15-year cartridge refueled machine delivering 400 MWth of heat at 800ºC core outlet temperature. An intermediate loop (of He or CO_2 or molten salt) carries the heat to a Ca-Br modified UT-3[1] water cracking cycle for the manufacture of H_2 (and O_2). The water cracking cycle rejects heat at 550ºC and that heat is used in a turbogenerator to provide hotel load electricity (and optionally to provide process heat). A multi-stage flash desalinization plant receives discharge heat at 125ºC and the brine is the ultimate heat rejection from the cascaded thermodynamic cycles.

Core design work has established the feasibility of a 15-year (or even 20-year) refueling interval in a natural circulation cooled core within a rail transportable sized vessel. Passive decay heat removal has been shown to be feasible, and work on passive safety response and passive load following are targeted for deveopment and look to be favorable. Work to establish passive safety/passive load follow features will facilitate removal of all safety functions from the balance-of-plant (BOP)-a desirable feature facilitating indigenous BOP construction for job creation in developing countries.

For hotel-load electricity production, a supercritical CO2 Brayton cycle turbogenerator operating on the reject heat (at 550ºC) from water cracking was evaluated during year 2. The evaluations have shown that this cycle holds strong potential for cost reduction compared with standard Rankine steam cycle options; for example, a turbine of no more than six stages, two compressors of one to three stages each, and a finned tube economizer should achieve 45 percent conversion efficiency. The big payoff is in BOP capital cost. A 400-MWth turbine is approximately 1 m long by 2 m diameter; a reduction factor of about 50 in footprint from Rankine steam cycle equipment. Additionally, a H_2/O_2 combustion gas turbine was considered as an on-site peaker plant for electricity production and a bank of stationary fuel cells will be evaluated as part of the project.

A detailed evaluation was made of the S-I and the UT-3 water cracking thermochemical cycles during year 1 and a modified version of the UT-3 process has been selected. The modifications raise the theoretical efficiency to approximately 65 percent and simplify the flow sheet, offering a potential to achieve practical efficiencies in the 45 to 50 percent range in a cost-effective engineerable process. In the second year, a flow sheet and preliminary heat balance have been worked out which replace the bromine regeneration of the original UT-3 flow sheet with a plasma cracking of HBr. A detailed program of research and development has also been devised and proposed to bring the Ca-Br process to a state of readiness for a demo prototype (under separate funding). Work on reaction

[1] University of Tokyo Process

kinetics and equipment sizing to support the larger effort will be a focus for year 3.

Silicon carbide is a candidate cladding for use in high-temperature, lead-cooled nuclear reactors, and composites using silicon carbide fiber for in-vessel components are also being considered. In year 2, a corrosion test was initiated to test the compatibility of silicon carbide in molten lead at elevated temperatures (650°C to 800°C). The test uses the quartz convection loop design (shown in Figure 1) previously used in tests of various steels with lead and lead-bismuth eutectic at 550°C. A silicon carbide tube about 6 inches long and 0.5 inches in diameter was placed in the hot leg of the test loop and maintained at about 800°C while the other vertical leg was kept at about 650°C, providing for slow circulation of the lead by natural convection. The system was maintained in this condition under very low oxygen conditions for 1,000 hours (~40 days). At the conclusion of the test period the SiC showed no evidence of attack and strong evidence of nonwettability of the SiC surface by 800ºC Pb (see Figure 2).

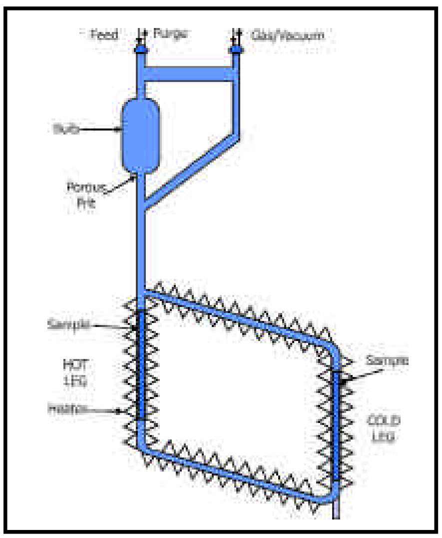


Figure 1. The schematic diagram shows the heavy metal corrosion test assembly.

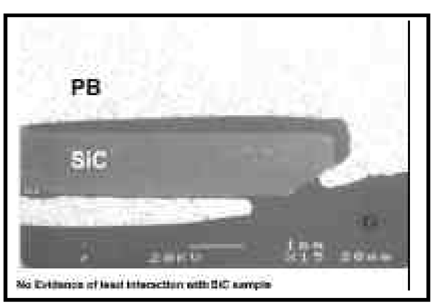


Figure 2. Experimental results show no evidence of lead interaction with the SiC sample.

Planned Activities

Although further optimization of core layout to flatten power density profile and increase average discharge burnup should be possible, the focus of reactor core design in year three will shift to design of thermostrucutral reactivity feedbacks to achieve passive load follow/passive safety. The heat balances of the several systems (reactor, heat transport, water cracking plant, SC-CO_2 cycle, and desalinization plant) will be integrated into an overall plant design; a load schedule will be developed consistent with the passive load follow/passive safety strategy, and a plant dynamics model will be created and used for safety analyses.

Contacts have been made with experts in the commercial optimization of desalinization equipment and with a U.S. utility already engaged in water desalinization. With their input, there are plans to evaluate hybrid options (flash evaporation/reverse osmosis) as alternatives to the current reference (flash evaporator) desalinization bottoming cycle.

Using corrosion test loop equipment emplaced in year 2, materials screening will be conducted in 800ºC Pb of additional structural material candidates (e.g., ZrC, ZrN, vanadium alloys, or others).

NUCLEAR ENERGY RESEARCH INITIATIVE

Development of Design Criteria for Fluid Induced Structural Vibrations in Steam Generators and Heat Exchangers

Primary Investigator: Ivan Catton, University of California, Los Angeles

Project Number: 00-062

Project Start Date: August 2000

Project End Date: September 2003

Research Objectives

Three objectives were defined for Phase II of this project whose goal is to develop degisn criteria for fluid induced structural vibrations in steam generators and heat exchanges:

(1) Single Phase Flow: To measure velocities and pressure distributions in square arrays with different pitch-to-diameter ratios for rigid and flexibly mounted tubes. Several heated tubes to be included to determine the relationship between instability avoidance and heat transfer penalty. Determine the entrance length and develop a stability map.

(2) Two Phase Flow: To determine the fluid-elastic instability conditions on tube arrays held at different orientations to the flow. Based on the data, develop an instability map. Conduct single tube steam-water flow and stability tests and compare results to air-water data. Measure the flow and when the tubes are heated to cause boiling. Establish the differences between air-water and steam-water flow and stability characteristics.

(3) Theoretical Development: Develop numerical algorithms for solving the governing equations using the vorticity transport equation as a first approximation. Use data from measurements of pressure differences in an array of rigid cylinders to check the validity of pressure variation assumption. Develop an energy equation for use in steam-water studies. Begin the development of models to describe the instabilities. Initiate development of tools to model the entrance region. Improve the constitutive relations and models needed for flow and stability modeling. Incorporate basic design data (tube material, diameter, wall thickness, length, type of support, and internal flow and heat transfer) into the models. Develop a relationship between tube supports used in experiments and prototypic supports.

The deliverables and milestones will include reporting experimentally based, single-phase entrance region flow and stability maps. Air-water stability maps will be developed and reported, and the impact of tube boiling will be reported. A decision about the level of detail needed in experimental studies of steam-water will be made. A report will be generated on the comparison of predictions with experimental results, and areas of weakness will be delineated; on the importance of prototype design variables in conducting laboratory tests of stability; and, on the results from evaluation of the impact of vibration avoidance on heat transfer.

Research Progress

Progress made on each objective will be reported in turn.

Single-phase Flow: The focus of the experiment has been shifted to the measurement of real-time, two-dimensional displacement of the tubes, rather than the measurement of the flow velocity field and pressures distributions around the tubes. The reasons for this shift is to focus attention on the primary result of instability, that of tube displacement. Further, tube displacement will be compared to model predictions. Though the eventual measurement of the velocity and pressure fields are essential, the top priority is to first validate the experimental design and construction by measuring the tube-displacement magnitudes for the range of velocities of which the system is capable. When fluid-elastic instability is achieved (as determined by the reduced velocity to vibration correlation) more detailed measurements can be made.

In designing and constructing a single-phase flow loop for supplying water at room temperature, the salient features of the loop include the following (see also Figure 1):

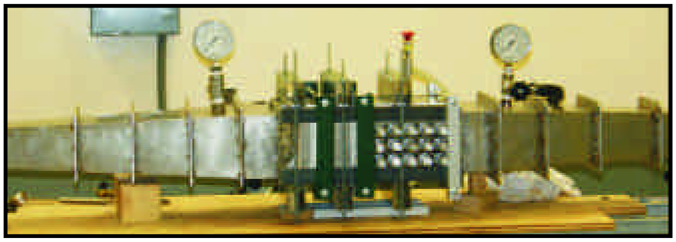


Figure 1. The photograph shows the single-phase test section and diffusers.

- A 15 hp centrifugal pump to supply 150 gpm of water at atmospheric temperatures
- A water accumulator ahead the test section in order to damp out the vibrations in the fluid
- A differential pressure flow meter

The experimental test section consists of the following:

- Inlet diffusers to avoid flow separation and flow straightener to make the flow uniform
- Acrylic box where a 5x3 dummy array of fixed tubes and a 5x3 array of flexible tubes are arranged in a normal square pattern
- Outlet diffuser
- External structure designed and installed to reinforce the test section and the diffuser against bending
- Pressure gages installed ahead and after the section

The flexible array of tubes consists of stainless steel tubes, 9 inches long and with a 1 inch outer diameter, suspended from the walls of the test section using stainless steel piano wires. A tensioning mechanism composed of high precision compression springs allows the user to change the tension in the piano wires and therefore the natural frequency of the tubes. The test section is optically accessible: a non-intrusive measurement system has been used to observe the tube motion.

The Data Acquisition System consists of the following:

- Two high-speed (120 frames per second) high-resolution (640 x 480) cameras
- Two computers (1.2 GHz processor, 1 Gigabyte RAM memory) dedicated to image acquisition

Preliminary results include the following:

(1) The test section has been assembled and tested.

(2) The array of flexible tubes has been calibrated, using one camera and a strobe lamp at a frequency of 16 Hz in still water (see Figure 2).

(3) Movies have been taken of the tubes at different flow rates.

(4) During the current development of methodology for data reduction, tresholding, filtering, subtraction, and cross-correlation of images techniques are being used to determine tubes displacements and orbits.

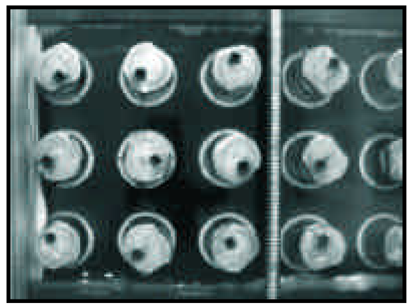


Figure 2. The photograph shows the camera output image with the array of vibrating tubes.

(5) Implementation of image processing codes has been accomplished using software such as Java Image, Adobe Photoshop and Matlab.

<u>Two-phase:</u> Design and construction of a two-phase flow loop for supplying both an air-water and steam-water mixture has been completed. The salient features of the loop include the following (see also Figure 3):

- A 10-hp centrifugal pump to supply 70 gpm of water at atmospheric pressures and temperatures
- A 46-kW electric boiler to generate steam

The experimental test section features a 5x3 dummy array of fixed tubes and a 5x3 array of flexible tubes arranged in a normal square pattern. The stainless steel

Figure 3. Pictured are the two-phase flow loop and test section.

tubes are 8 inches long and have an outer diameter of 0.625 inches, and are suspended from the walls of the test section using stainless steel piano wires. Strain gages mounted on shims welded to the piano wire are used to measure the natural frequency and damping coefficient of the tubes as they vibrate. A half bridge configuration of strain gages is used to compensate for change in length of the piano wire due to temperature.

Preliminary results of the two-phase flow include the following:

(1) The strain gages have been successfully used to calculate the natural frequency of vibration of the tubes in air and water, as well as the damping coefficient (ζ).

(2) These two quantities have been calculated from the time history of the strain gage signal and also the Fourier transform of the signal.

(3) The natural frequency of the tubes in air is found to lie between 20-25 Hz.

(4) A damping coefficient of 1 percent is observed in air.

(5) Unstable motion of the tube characterized by large amplitude vibrations beyond a particular value of the air flow rate has been observed in air-water flow.

(6) Movies have been made of the instability. The strain gage signal also shows the instability as shown Figure 4.

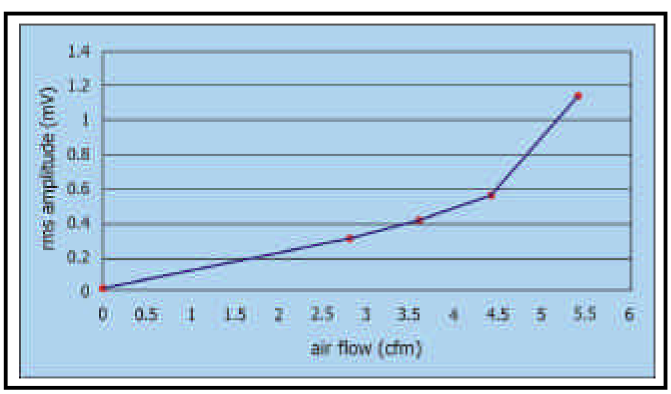


Figure 4. The graph illustrates fluid-elastic instability in air-water flow (water flow rate held constant at 32 gpm).

Planned Activities

For the single-phase effort, more experiments will be conducted with the existing experimental setup to determine the relationship between the RMS displacements of each of the tubes in the array as a function of the flow speed, so that instability maps can be drawn. The image-processing program will be developed and used for preliminary analysis of the experimental data; using cross-correlation techniques between subsequent images, it will be possible to measure the tube displacements as well the direction of motion. Additional properties of the tube vibrations, such as relative tube motions, orbits followed by the tubes, and statistical properties, will be computed from the displacement data.

In the two-phase experiment, after successful demonstration of fluid-elastic instability using a single tube in two-phase flow, the entire 15-tube array is being installed in the test section. Once this is done, baseline tests will be made for the instability in single phase flow. The results of this test will be used to determine the parameters required for the two-phase tests. Fluid-elastic instability results in air-water and steam-water flow will be obtained. An effort will also be made to determine the damping characteristics of the tube in two-phase flow. Following this, tests using a rotated square array of tubes will be carried out.

Additionally, papers have been presented and abstracts have been or will be submitted for several conferences in 2002 and 2003.

NUCLEAR ENERGY RESEARCH INITIATIVE

An In-Core Power Deposition and Fuel Thermal Environmental Monitor for Long-Lived Reactor Cores

Primary Investigator: Don W. Miller, The Ohio State University

Collaborators: University of Akron; Westinghouse Electric Company

Project Number: 00-069

Project Start Date: July 2000

Project End Date: September 2003

Research Objectives

The primary objective of this program is to develop the Constant Temperature Power Sensor (CTPS) as an in-core instrumentation that will provide a detailed map of local nuclear power deposition and coolant thermal-hydraulic conditions during the entire life of the core. In the case of the some DOE Generation IV reactor cores, this could include normal operation, post-accident operation, and monitoring after the core is placed in permanent storage. The sensors used in this instrumentation must have a lifetime comparable to the core and be compatible with the neutronic and thermal conditions expected over the range of proposed Generation IV reactor designs. Further, the sensors must be robust and capable of operation even with extensive material degradation and, if required to achieve this objective, they must provide for in situ calibration and performance monitoring.

The CTPS concept is based on the idea of maintaining the temperature of a small mass of actual reactor fuel or fuel analogue constant by adding heat through resistive dissipation of input electrical energy. A feedback control loop is used to provide the exact amount of input electrical energy needed to keep the fuel mass at a specified constant temperature, well above the coolant bulk temperature, regardless of the nuclear energy deposited in the mass. Energy addition to the fuel mass and fuel temperature feedback to the controller are both provided by a simple resistive heating element embedded in the fuel mass. The input electrical energy required to maintain a constant temperature provides a measure of the actual nuclear energy deposition, since they are inversely related.

Research Progress

This report provides a summary of the progress made on the tasks specified for the second year of the three-year program as described in the program proposal. The remainder of this section is a short narrative that summarizes progress, key accomplishments, and significant problems encountered, and their resolution, for each of the second-year tasks.

Second Year Program Tasks: The Ohio State University

- The task, "Modification of the Numerical Model to Represent the Planar Constant Heat Flux Power Sensor (CHFPS)," was originally unplanned, but became required when OSU investigators decided to fabricate and evaluate planar screen-printed sensors. This task was completed and results are being used in the design of the planar CHFPS. (A previous task, completed in Year 1, involved modification of the numerical model to represent the cylindrical CHFPS.)
- As noted in the last Annual Report, it was substantially more challenging than expected to design and fabricate sensor prototypes that will reliably operate in the expected environment. Initially, an attempt was made to utilize a method comparable to the first prototype. However, it soon became apparent that due to both size and required operational temperature this may not have resulted in a sensor with the required performance characteristics. Consequently, a sensor was designed with a planar configuration that requires a controlled thin layer of platinum on a sensor cone comprised of uranium oxide. To obtain a layer sufficiently robust to permit the sensor to reliably operate in the expected environment has required significant development of new materials technology.

- Good progress has been made on developing the planar configuration, and based on preliminary testing, investigators believe it will be sufficiently robust to operate in the environmental conditions of the International Reactor, Innovative, and Secure (IRIS) concept. To validate this conclusion, the numerical model was modified to represent this configuration and it has been used for design and performance modeling. Simultaneous progress was made on the original cylindrical configuration and it is now believed that both configurations will have a high probability to operate successfully in the challenging environmental conditions posed by the IRIS. Operational prototypes of both configurations are expected by the end of August.
- A critical test during the fabrication process of both sensors is evaluation of the physical integrity of components and contacts at temperatures above expected operational temperatures, therefore providing a preliminary test of sensor materials.
- As discussed in the Annual Report, the Test Digital Controller and Thermal Monitoring System was extensively tested in the first year. It will continue to be tested during completion of second-year tasks.
- In the task to develop and evaluate in-situ performance monitoring, both compensation and fault detection will be used to identify any discrepancies between the actual sensor and a sensor model. The process of compensation uses the sensor output and the sensor model to estimate the environmental temperature and the heat transfer coefficient. Fault detection will be used to obtain current sensor characteristics by measuring both the sensor input and output. A difference between the measured dynamics and the model indicates a fault. This protocol has been evaluated with the numerical model and will be used as the basis for an on-line monitor of CTPS performance, which will be evaluated experimentally during the third year of the program.
- One of the first tests of both the planar and cylindrical sensors will be a short-term radiation exposure that provides a preliminary radiation test of sensor materials. As discussed in the Annual Report for the first year, a potential problem was identified related to long-term testing, which was to begin in the second year. In preparation for this task, OSU investigators had several discussions with the staff at the Ford Research Reactor at the University of Michigan and determined that this facility will meet program needs. However, it should be noted that the Ford reactor is scheduled for permanent shutdown, pending a review by the Department of Energy for additional funding.

Second Year Program Task: Westinghouse Electric Company

- Investigators at Westinghouse have started work on the task to evaluate the effects of the sensors on the reactor environment. The selection of the first reference core configuration for IRIS is nearing completion. The sensor dimensions, bulk materials, and quantities and isotopic mixtures of fissionable materials will be used with the currently available information on IRIS core configuration to evaluate the perturbations on the neutron and temperature environment resulting from the presence of CTPS and CHFPS units. This task will be completed by the end of August 2002.
- The Year 2 milestones of completing characterized prototype sensors and controllers, and identifying effects of in-core monitors on the reactor designs will be completed by the end of the second month of the third year. Although this is two months behind the planned schedule, the Westinghouse research program is in a good position to complete the program as planned. More importantly, a sensor configuration—a planar sensor with screened printed electrodes—will be available that is likely to be superior to the planned cylindrical sensor configuration.

Planned Activities

The following milestones have been established for the third year of the program. Selected activities to attain each milestone are described after each respective milestone. A detailed test plan is currently being developed.

(1) Complete long-term irradiation and thermal cycle testing. Cylindrical and planar sensors will be placed in the University of Michigan Ford Research Reactor in a location with maximum fluence over an approximate twelve-month time period. The physical integrity and electrical output of the sensors will be monitored.

(2) Complete numerical model validation. A static calibration will be completed of cylindrical and planar sensors over as wide a dynamic range as possible in the OSU Research Reactor. Additionally, a dynamic calibration of cylindrical and planar sensors will be completed over as wide a dynamic range as possible in the OSU Research Reactor. This will require measurement of the sensors' transfer functions using a local flux oscillator and neutron noise analysis, scram response, and response, following various positive reactivity insertions. Calibrations and testing will take place over a range of temperatures from ambient through 800°C.

(3) Evaluate in situ calibration algorithms. Performance evaluation will take place during numerical model validation and other operational scenarios.

NUCLEAR ENERGY RESEARCH INITIATIVE

Design and Construction of a Prototype Advanced On-Line Fuel Burn-up Monitoring System for the Modular Pebble Bed Reactor

Primary Investigator: Bingjing Su, University of Cincinnati

Collaborators: North Carolina State University; Massachusetts Institute of Technology (MIT)

Project Number: 00-100

Project Start Date: August 2000

Project End Date: September 2003

Research Objectives

The overall goal of the project is to conceptually design, construct, and test an advanced on-line fuel burn-up monitoring system for the next-generation Modular Pebble Bed Reactor (MPBR). The MPBR is a high-temperature, gas-cooled nuclear power reactor currently under study as a possible Generation IV system. In addition to its inherently safe design, a unique feature of this reactor is its multi-pass fuel cycle in which the graphite fuel pebbles are randomly loaded and continuously cycled through the core until they reach their prescribed End-of-Life burn-up limit (~80,000 MWD/MTU). Therefore, an on-line measurement system is needed to accurately assess whether a given pebble has reached its End-of-Life burn-up limit and thereby provide an on-line, automated go/no-go decision on fuel disposition on a pebble-by-pebble basis.

This project investigates approaches to analyzing pebble bed fuel in real time using gamma spectroscopy and possibly using passive neutron counting of spontaneous fission neutrons to provide the speed, accuracy, and burn-up range required for the MPBR. The project involves all phases necessary to develop and construct a burn-up monitor, including a review of the design requirements of the system; identification of materials and methodologies that would satisfy the design requirements; modeling and development of potential designs; and finally, the construction and testing of an operational detector system.

Research Progress

The project is to be conducted in three phases. Phase 1 is designed to characterize the fuel burn-up and radiation of fission products and has three specific tasks:

A) Identification of the system requirements
B) Fuel depletion modeling
C) Radiation/burn-up correlation analysis

The objective of Task A is to determine the design requirements for an on-line burn-up monitoring system to be used with the MPBR plants. This task involves collecting the most recent design parameters from the MPBR design effort underway at MIT, the Idaho National Engineering and Environmental Laboratory, and South Africa's ESKOM[1]. Based upon the information collected, the required availability of the burn-up monitor, pebble throughput rates, burn-up ranges of interest, cool down time for pebbles prior to measurement, and other basic physical parameters that are needed for the on-line burn-up monitor design have been determined. The objectives of Tasks B and C are to accurately model the buildup of fission products and transuranic elements in irradiated fuel pebbles and to identify correlations between types, amounts, and spectra of radiation emitted from a fuel pebble and the fuel burn-up level of the pebble. To this end, the ORIGEN2.1 code was used to perform the fuel depletion calculations and the utilization of passive gamma-ray spectrometry and neutron counting methods was investigated to establish a power-history and cooling-time insensitive burn-up measurement approach, which relies on the relationship between the burn-up and the radiation emitted by the fuel pebble from fission products and heavy actinides that have built up during its passing through the core.

In the case of gamma-ray measurements, it was found that accurate and predictable correlations between activity and burn-up can result if Cs-137 (~30.2 years half-life) and/or Eu-154 (~8.5 years half-life) are used as indicators. The correlation is linear in the case of Cs-137

[1] Electricity Supply Commission

and fourth order in the case of Eu-154. The activity of these radionuclides exhibited a behavior that is highly independent of the cooling time and the in-core power history (i.e., the pebble's path in the core), which resulted in variations in the predicted discharge burn-up of approximately 3 to 5 percent when the cooling time was varied between 0 and 7 days and the power was varied between 50 percent and 150 percent of the nominal thermal core power. However, for Cs-137 the activity measurement will be performed using the 662 keV gamma line, which is significantly interfered by the 658 keV line of Nb-97. This spectral interference cannot be completely resolved even when using germanium high-resolution detectors.

In addition, the option of using artificially introduced dopants (e.g., Co) as burn-up monitors was studied. The use of such dopants may be desirable because they can provide intense and high-energy gamma rays that have improved signal to noise characteristics, and therefore, can improve the accuracy of the measurement. Results indicate that the relative activity ratio of Cs-134 to Co-60 is somewhat more resistant to power variation and thus is a potential indicator of discharge burn-up that is accurate to within 5 percent. Furthermore, the use of a relative burn-up indicator may eliminate the need for an absolute efficiency calibration of the gamma ray detector and thus minimize the contribution of the detector calibration error in the final uncertainty analysis of the burn-up monitoring system.

The use of passive neutron counting has also been investigated as a method for burn-up measurement. In this case, the neutrons are generated either by the spontaneous fission of heavy actinides or from the (α,n) reactions that take place within the pebble. The depletion calculations indicate that the (α,n) component is negligible compared to the contribution of spontaneous fission. Moreover, the spontaneous fission component is dominated by the contribution from Cm-244 that has a half-life of 18.1 years. This allowed the establishment of a correlation between the total number of neutrons emitted by a pebble and its calculated burn-up, using a fourth order polynomial. Moreover, the total number of neutrons emitted is substantially insensitive to the cooling time and the in-core power history, which resulted in variations in the discharge burn-up prediction of less than 10 percent when the cooling time varies between 0 and 7 days and the power varies between 50 percent and 150 percent of the nominal thermal core power.

Although the work of Phase 1 concludes that passive gamma-ray spectrometry of selected fission products and passive total neutron counting have the potential to be developed as credible methods for on-line burn-up measurement, the feasibility of using any of the above approaches must take into account other considerations. Paramount among these is the ability to perform the measurements within the realistic requirements of throughput and the overall system reliability. That is the focus of Phase 2 efforts, with the objectives of establishing signal-to-source response functions for candidate detector systems, detector selection and optimization, and conceptual system design. To date, the following has been accomplished:

(1) The signal-to-source response functions have been simulated by using MCNP for several candidate gamma ray spectrometers that are based on either cryogenically cooled or room temperature HPGe detectors. A typical result is shown in Figure 1. In addition, the Gaussian broadened spectra for these detector responses are produced by using the FWHM vs. Energy relation generated by SYNTH.

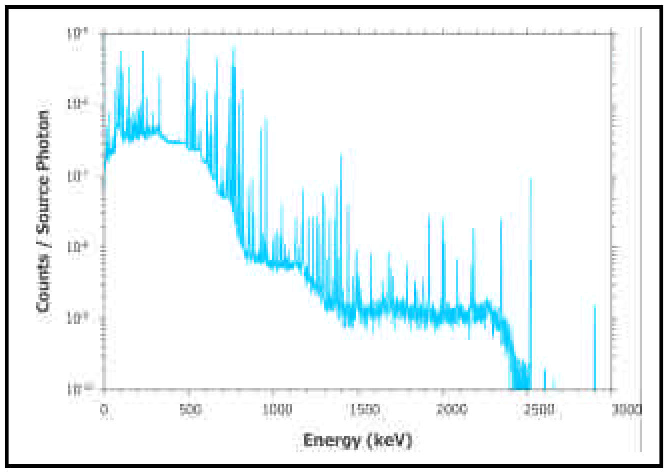


Figure 1. MCNP simulated gamma spectrum from a pebble at a burn-up of 80,000 MWD/MTU

(2) The minimum detectable activities of candidate fission products were calculated for gamma measurements. The results indicate that for the gamma spectroscopy, with approximate values for the detection efficiency, the expected measurement time that would allow a statistically accurate result for burn-up measurement varies between 30 to 60 seconds. This will be sufficient to meet a circulation rate of one pebble every 30 seconds, especially if a multi-detector system with enhanced detection efficiency is used.

(3) Detection efficiency for neutron counting has been simulated by MCNP for several common neutron detectors, including gas recoil counters, fission chambers, and passive neutron assay by the (n, γ) conversion. Due to the low emission rate of spontaneous fission neutrons, only the BF3 long tube counter is found to be able to register a statistically meaningful neutron count within 30 to 60 seconds, if all background noises are neglected.

(4) The interference from gamma rays to a BF3 neutron counter was also simulated by MCNP. Results show that pulses (in 2 to 3 MeV) produced by gamma rays in the detector overwhelm the neutron pulses, due to the extremely high ratio ($> 10^9$) of gamma emission rate over neutron emission rate. Such interference cannot be overcome by using gamma shielding. Although other neutron detection schemes will be investigated in the next phase, it seems unlikely to have an accurate neutron counting within the required time interval, considering the interference from gammas. Therefore, it is unlikely that neutron counting is usable for on-line burn-up determination.

Planned Activities

The research efforts in Phase 2 involved developing the signal-to-source response functions for candidate detector systems and identifying optional detection methods for use in the burn-up monitor. The objective in the remaining phase of this project is to finalize an optional (hopefully optimal) design for the on-line, burn-up monitor system and to test the functionality and performance of the system by laboratory experiments. To this end, the following research activities will be carried out:

- Investigating other neutron detection methods to make a conclusion about the feasibility of using neutron counting for on-line burn-up determination
- Assessing the summing effect on gamma ray spectra in gamma measurement
- Studying further the feasibility of Co-doping for burn-up measurement
- Constructing a high resolution gamma spectrometry detector system as a burn-up monitor
- Conducting several experiments at the North Carolina State University's PULSTAR Reactor Laboratory to evaluate fundamental gamma spectrometry issues that may affect the burn-up measurement

NUCLEAR ENERGY RESEARCH INITIATIVE

Balance of Plant System Analysis and Component Design of Turbo-Machinery for High-Temperature Gas Reactor Systems

Primary Investigator: Ronald G. Ballinger, Massachusetts Institute of Technology

Collaborators: Northern Engineering & Research

Project Number: 00-105

Project Start Date: August 1999

Project End Date: September 2003

Research Objectives

The purpose of this project is to develop systems analysis tools for the evaluation of turbo-machinery and balance of plant (BOP) power conversion in high-temperature gas cooled reactor (HTGR) systems. These tools will then be used to develop optimized power conversion systems for HTGR systems. Current concepts for HTGR systems call for modular designs with electrical output in the 110-MWe range. Key questions which must be addressed in order for such systems to be adequately evaluated include:

(1) Can a helium power turbine be developed in the 110-MWe range?

(2) Can advanced compact heat exchanger technology be used in the design of intermediate heat exchangers (for indirect cycle plants) and/or recuperators (direct and indirect cycle plants)?

(3) Can structural and materials issues be adequately characterized to allow for detailed life-cycle analysis?

(4) How do specific component designs impact overall cost?

Research Progress

Key progress to date includes the following:

(1) A "reference" plant design was established, which can be built with existing technology. However, the thermal efficiency of the plant will be sub-optimal due to restrictions on the reactor outlet temperature.

(2) Initial designs for the turbo machinery and heat exchangers were established, including the intermediate heat exchanger. Two concepts have been identified. However, each design will require that active measures be taken to assure safe operation.

(3) Potential vendors for all of the major components were identified and initial cost estimates for these components were obtained.

(4) A steady state model was developed for the overall plant and is being used to help optimize the system configuration.

(5) The initial transient model was developed and the process of refining the individual component models is underway.

(6) A reactor model was developed for use in both the steady state and transient models.

Each of these points is discussed in more detail below.

Project Design Goals

The overall design goals of the project were established as follows:

(1) To develop a "reference" plant design that could be built today with no significant advance in technology. This design must satisfy all appropriate codes and standards and should not require a significant R&D effort on the part of the component vendors.

(2) To answer key questions related to the design of the power conversion system. These questions encompass three general categories of (1) intermediate heat exchanger design, (2) turbine and compressor design, and (3) system control.

(3) To then develop an "advanced" design which allows for prudent and achievable extensions of

technology. In this context, "prudent and achievable" means that some development effort may be needed but that this development effort should not be a significant fraction (< 20 percent) of the cost of the system.

The design team also had to make an assumption related to the nuclear island. While it was not the task of the project to "design" the nuclear island, it was necessary to choose a reference system as the starting point. For this project, researchers elected to choose the current ESKOM PBMR reactor as the reference. They believe that this reactor is a reasonable choice because the design fits with a Brayton cycle plant that is currently being designed for commercial use and therefore will be constrained by the same codes and standards as the BOP design.

The foregoing design goals imply certain constraints on the design process. The general design constraints for the design have been established as follows:

- Compliance with the ASME Boiler and Pressure Vessel Code, Section III, Class I for the primary pressure boundary which includes the reactor vessel and piping and the primary side of the intermediate heat exchanger
- Compliance with the ASME Boiler and Pressure Vessel Code, Section VIII where applicable
- Purchasable components
- Use of the ESKOM PBMR as the primary heat source

<u>ASME Code Compliance</u>

ASME code compliance places limits on temperature, allowable stress, and materials selection. Section III qualified materials are the most severely limited in the entire ASME code. The requirement of Section III results in an intermediate heat exchanger inlet temperature of 400°C. In addition to this temperature limit, the maximum allowable stresses at these temperatures place severe design constraints on the intermediate heat exchanger and will require that an "operating" curve for this component be established in which time and temperature for a given differential pressure (primary to secondary) accumulation are measured and accumulated to predict component life and assure safe operation. As a practical matter this means that for certain accident scenarios, one or both of the primary and secondary sides will need to be pressure-regulated.

<u>Purchasable Components</u>

This design constraint results in constraints due to technology limits. For the initial phase of the project this meant that there was a limit for the maximum shaft power to approximately 50Mw (70,000 shaft HP). This limit translates into limitations on the minimum number of shafts in the BOP system. This in turn will have a direct, and negative, effect on the complexity of the control system for the BOP. For the initial design this translated into a 4 shaft BOP with three compressor/turbine sets and a two-set power turbine design.

A second limit concerned the maximum allowable helium velocity. The establishment of this limit was more subjective and the team relied on discussions with various turbine and heat exchanger vendors. The graphite structure in the PBMR core results in a velocity limit of 50 m/sec due to erosion limitations. However, with the use of an indirect cycle, this limitation does not exist. Additionally, there are no sonic limitations with helium. However, based on discussions with component manufacturers, an upper velocity limit of 120 m/sec was established. This velocity limit, for a fixed system mass flow rate and temperature, places constraints on piping diameters.

<u>ESKOM Reactor Limits</u>

The use of the ESKOM PBMR reactor as the primary heat source carried with it all of the materials limitations for this plant. These include the consequences of using a conventional A 508 steel for the pressure vessel and piping. The design also resulted in geometric constraints related to piping configuration and layout.

The use of A 508 steel for the pressure vessel places limits on temperature of 280°C for steady state operation and 350°C for transient, short time, operation. Due to the high gas outlet temperature (850°C in the initial design), this requires that an active cooling system be in place during operation. For the chosen design, this also means that the intermediate heat exchanger (IHX) must have a cooled Section III boundary. Additionally, since the source of cool gas for the ESKOM design is the compressor outlet, the use of an IHX complicates the design. The design "fix" for this constraint is to include, as either a separate heat exchanger or as a part of the IHX, the capability for using a compressor outlet gas stream to cool a separate cooling circuit for the pressure vessel and associated piping. As a practical matter this means that, due to the piping cooling requirement, the hot sections of the BOP (to the turbine inlets) must also have cooled pressure boundaries.

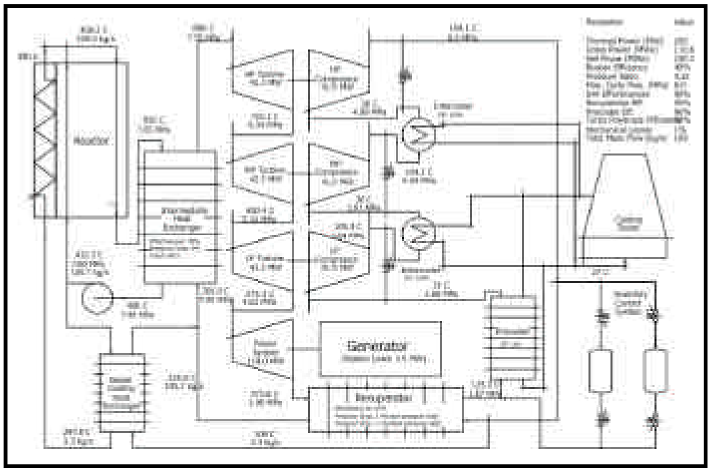


Figure 1. The schematic illustrates the reference BOP System.

Current Reference Plant Design

Given the initial design requirements and constraints, a reference plant has been designed and configured. Figure 1 shows a schematic of this design. Temperatures, pressures, and flow rates have been identified.

Key features of the reference system include the following:

- Use of a secondary heat exchanger for the reactor vessel and piping cooling system
- Use of both inventory and bypass control (to be discussed further below) for system control
- Location of bypass control valves on the cold side of the plant
- Use of intercooling

The reference plant is a 250 MWth design with a gross power of 110.6 MWe, which yields a net efficiency of 40 percent. The net plant efficiency is considerably less than the overall project goal of 45 percent. However, it must be cautioned that the initial phase of the program was to specify a plant with only the potential to be built using existing technology. An advanced reference design will be developed during the second year of the project that will seek to increase the overall efficiency.

Planned Activities

The first year of the project has seen the development of the reference design, and researchers are satisfied that the design could be built, although its estimated efficiency is below the target efficiency for an economic design. An increase in reactor outlet temperature of from 850°C to 900°C will be needed to achieve this efficiency. Designs for major individual components have also been taken to the point where the team is satisfied that they could be built. In the cases of the turbines, compressors, and the IHX, it was possible to obtain initial cost estimates from commercial sources and it is believed these components could be purchased with a minimal requirement for extensive R&D effort.

The next steps in the project will be to more clearly establish the limits to the achievement of an increase in temperature. Based on this analysis, a final system design will be developed.

Although not mentioned previously in this report, work has begun on the development of a transient model during this reporting period. During the next year, the transient model will be fully developed and then used in optimization of the final design.

NUCLEAR ENERGY RESEARCH INITIATIVE

Forewarning of Failure in Critical Equipment at Next-Generation Nuclear Power Plants

Primary Investigator: Lee M. Hively, Oak Ridge National Laboratory

Collaborators: Duke Engineering and Services Inc.; Pennsylvania State University

Project Number: 00-109

Project Start Date: August 2000

Project End Date: September 2003

Research Objectives

Researchers at Oak Ridge National Laboratory (ORNL) are applying new nonlinear methods to assess condition change and forewarn of machine failures from experimental test sequences. These new measures are much more discriminating than those previously applied. The specific application is critical equipment in next-generation nuclear power plants. Test data have been provided by two collaborating institutions, namely Duke Engineering and Services during fiscal year (FY) 2001, and the Pennsylvania State University during FY 2002. If successful, this effort will overcome one of the major present hurdles to automatic, timely, and reliable prognostication for condition-based maintenance and repair.

The methodology is multi-tiered, model-independent, and data-driven. The first tier rejects data of inadequate quality. The second tier removes signal artifacts that would confound the analysis, but leaves the relevant nonlinear dynamics essentially unaffected. The third tier converts the artifact-filtered, time-serial data into a phase-space (geometric) representation, which then is transformed to a discrete distribution function (DF). The nominal-state DF is compared to subsequent test-state DFs via dissimilarity measures of condition change, which are much more sensitive than conventional statistics or traditional nonlinear measures. Thus, the discriminating power of the method is less affected by noisy, finite-length datasets.

ORNL's approach yields robust nonlinear signatures of degradation, allowing earlier and more accurate detection of the deterioration onset, as well as more accurate predictions of the progress of the deterioration. Anticipation of failures in critical equipment at next-generation nuclear power plants will help in the scheduling of maintenance to minimize safety concerns, unscheduled non-productive downtime, and collateral damage due to unexpected failures. It is expected that this approach will lead to significant economic benefits and improved public acceptance of nuclear power.

Research Progress

Long-term failure monitoring of operational equipment is not feasible within the scope of our present project, since such failures may take years to occur. Instead, data was acquired in FY 2001 from a motor-driven pump for two test sequences of progressively larger seeded faults (imbalance and misalignment). ORNL's nonlinear measures of condition change correlated well with the experimental level of vibration, both below and above the ISO 2372 and ISO 3945 limits. This work included a robust implementation of the nonlinear analysis on a desktop computer, not unlike that for acquisition and analysis at an advanced nuclear reactor. The Annual

Data Provider	Equipment and Type of Failure	Diagnostic Data
1) EPRI (S)	800-HP electric motor: airgap offset	motor power
2) EPRI (S)	800-HP electric motor: broken rotor	motor power
3) EPRI (S)	500-HP electric motor: turn-to-turn short	motor power
4) Otero/Spain (S)	¼-HP electric motor: imbalance	1D acceleration
5) PSU/ARL (A)	30-HP motor: overloaded gearbox	load torque
6) PSU/ARL (A)	30-HP motor: overloaded gearbox	accelerometer power
7) PSU/ARL (A)	30-HP motor: overloaded gearbox	accelerometer power
8) PSU/ARL (S)	crack in rotating blade	motor power

Table 1: Summary of PY2 Test Sequences. Table 1 also shows the type of diagnostic data that was analyzed for failure forewarning. Motor power, P, was obtained from the three-phase motor currents, I_i, and voltages, V_i, by using the formula, $P = \; i I_i V_i$. The sum over i includes all three electrical phases. Accelerometer power came from tri-axial acceleration, which is a three-dimensional vector, A, that can be integrated once in time to give velocity vector, $V = \; A\,dt$. Mass, m, times acceleration (vector) is force (vector), F = mA. The vector dot-product of force (vector) and velocity (vector) is power (scalar), P = F " V. The resultant three-dimensional accelerometer power captures the dynamics from all three components of acceleration.

Report, ORNL/TM-2001/195, provides details of the first year's progress, including validation of the approach on well-characterized model data.

Additional test data was obtained during the second project year, as summarized in Table 1. Some of the test sequences involved seeded faults (denoted by 'S' in Table 1), with the equipment initially in nominal operation, and subsequently with successively larger (controlled) faults, finally yielding equipment failure. A second class of accelerated failure tests (denoted by 'A' in Table 1) likewise began with nominal operation. The over-stressed equipment experienced a gradual (uncontrolled) degradation, and ultimately failed. For example, the gearbox failed by the breakage of one or more gear teeth.

ORNL's patented nonlinear measures showed clear change in every sequence, as the test progressed from nominal operation, through degradation to failure. Forewarning was not consistently seen in the conventional statistical measures, nor do traditional nonlinear measures give reliable forewarning. Figure 1 shows an example of the change in phase-space dissimilarity measures for Test-Sequence #2. This sequence began with the motor running in its nominal state. Then, one rotor bar cross section was cut 50 percent at the 11 o'clock position. Next, the same rotor bar cut completely through. Subsequently, a second rotor bar was cut 100 percent at the 5 o'clock position, and in addition to, the first rotor failure. Finally, two additional rotor bars were cut adjacent to the original 11 o'clock bar, with one bar cut on each side of the original, yielding four bars completely open. Figure 1 shows a linear rise in the logarithm of the phase-space dissimilarity measures of condition change, corresponding to the exponential rise in the magnitude of these seeded faults (successively doubling from 1/2 to 1 to 2 to 4). Research on the accelerated failure test sequences has yielded a statistical criterion that distinguishes between the gradual rise in phase-space dissimilarity measures and the abrupt (additional) increase that gives forewarning of failure. These results provide compelling evidence for failure prognostication via the ORNL nonlinear paradigm. Complete details are provided in the FY 2002 Annual Report, ORNL/TM-2002/183.

Other work during FY 2002 included actions to protect the intellectual property for this technology. Researchers responded to the U.S. Patent Office action on an earlier patent application that resulted in the allowance of all of

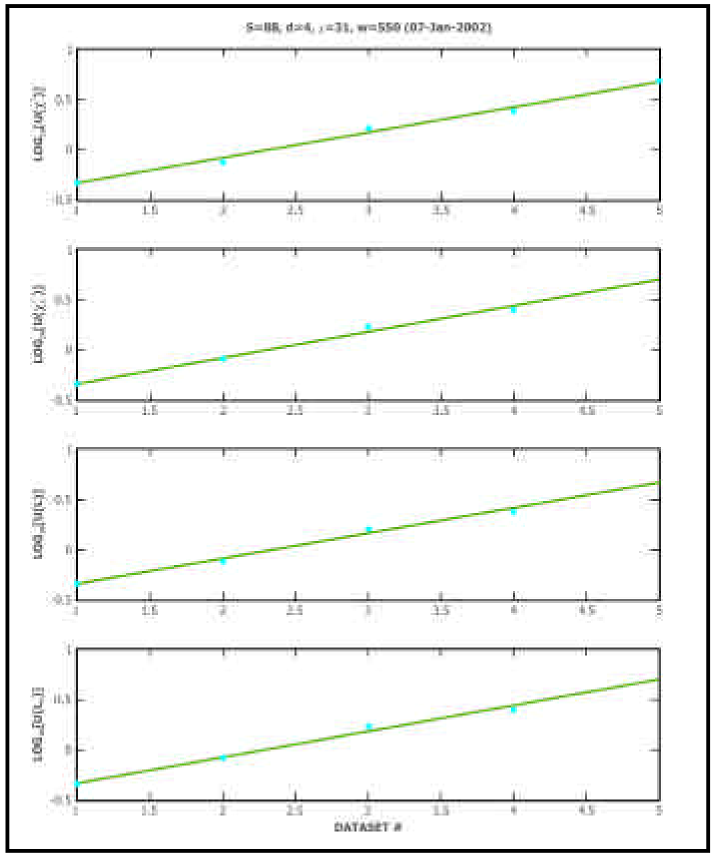


Figure 1. Plots of the four nonlinear dissimilarity measures for the broken-rotor seeded-fault power data. Dataset #1 is for the nominal (no fault) state. Dataset #2 is for the 50% cut in one rotor bar. Dataset #3 is for the 100% cut in one rotor bar. Dataset #4 is for two cut rotor bars. Dataset #5 is for four cut rotor bars. The exponential rise in the severity of the seeded faults is shown as an almost linear rise (solid line) in the logarithm of all four dissimilarity measures (*) for the chosen set of phase-space parameters.

the claims for the connected phase-space dissimilarity method. A new patent application was also submitted to protect additional improvements to the methodology that was developed more recently.

Planned Activities

Phase 2 work during the third project year (FY 2003) will involve acquiring and analyzing additional data from test sequences for seeded faults and accelerated failures. These tests are expected to include additional gearbox failures, cracked-shaft failures, and various generator failures in the rotor, stator, and diode. A companion effort will seek better discrimination and robustness of the forewarning methodology, along with a reduction in the analyst-intensive effort. Phase 3 work during FY 2003 will assess the usefulness of this forewarning approach in terms of failure reductions and of cost-effectiveness.

NUCLEAR ENERGY RESEARCH INITIATIVE

Feasibility Study of Supercritical Light Water Cooled Fast Reactors for Actinide Burning and Electric Power Production

Primary Investigator: Philip E. MacDonald, Idaho National Engineering and Environmental Laboratory (INEEL)

Collaborators: Massachusetts Institute of Technology (MIT); University of Michigan; Westinghouse Electric Corporation

Project Number: 01-001

Project Start Date: August 2001

Project End Date: September 2004

Research Objectives

The use of light water at supercritical pressures as the coolant in a nuclear reactor offers the potential for considerable plant simplification and consequent reduction in capital and operating and maintenance (O&M) costs compared with current light water reactor (LWR) designs. Also, given the thermodynamic conditions of the coolant at the core outlet (i.e., temperature and pressure beyond the water critical point), very high thermal efficiencies are possible for the power conversion cycle (i.e., up to about 45 percent). Because no change of phase occurs in the core, the need for steam separators and dryers as well as for BWR-type recirculation pumps is eliminated, which, for a given reactor power, results in a substantially shorter reactor vessel and smaller containment building than the current BWRs. Furthermore, in a direct cycle, the steam generators are not needed. If no additional moderator is added to the fuel rod lattice, it is possible to attain fast neutron energy spectrum conditions in a supercritical water-cooled reactor (SCWR). This type of core can make use of either fertile or fertile-free fuel and retain a hard spectrum to effectively burn plutonium and minor actinides from LWR spent fuel while efficiently generating electricity. One can also add moderation and design a thermal spectrum SCWR that can also burn actinides.

The project is organized into three tasks:

<u>Task 1 - Fuel-cycle Neutronic Analysis and Reactor Core Design (INEEL):</u> For the fast-spectrum SCWR, metallic (dispersion type), and oxide, fertile fuels will be investigated to evaluate the void and Doppler reactivity coefficients, actinide burn rate, and reactivity swing throughout the irradiation cycle. For the thermal-spectrum SCWR, a variety of fuel and moderator types will be assessed.

<u>Task 2 - Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking (University of Michigan and MIT):</u> MIT will use an existing supercritical-water loop to conduct corrosion experiments in flowing supercritical water. To collect stress-corrosion cracking data, a high-temperature autoclave containing a mechanical test device will be built at the University of Michigan in Year 1 and operated in Years 2 and 3. The data from both universities will be used to identify promising structural and fuel cladding materials and develop appropriate corrosion and stress corrosion cracking correlations.

<u>Task 3 - Plant Engineering and Reactor Safety Analysis (Westinghouse and INEEL):</u> The optimal configuration of the power conversion cycle will be identified. Particular emphasis will be given to the applicability of current supercritical fossil-fired plant technology and experience to a direct-cycle nuclear system. A steady-state, sub-channel analysis of the reactor core will be undertaken with the goal of establishing power limits and safety margins under normal operating conditions. In addition, the reactor's susceptibility to coupled neutronic/thermal-hydraulic oscillations will be evaluated. The response of the plant to accident situations and anticipated transients without a scram will also be assessed.

Research Progress

<u>Task 1 - Neutronic Analysis and Reactor Core Design:</u> A qualitative analysis was performed to determine which fuel form would support the highest reactivity-limited burnup in a fast-spectrum SCWR, and would have the most proliferation-resistant isotopics at a particular burn-up. A relatively long core life and a modest reactivity swing are possible in fast-spectrum SCWRs with most fuels. However, the uranium-based fuel types had the highest beginning-of-life reactivity, and the best reactivity-limited

burn-up, whereas the thorium-based fuels had the best spent-fuel isotopics. Therefore, the most appropriate fuel for fast-spectrum SCWRs appears to be a mixture of thorium and uranium to balance long core life with proliferation-resistant isotopics.

In addition, the neutronic performance of several solid moderators for use in a thermal spectrum SCWR core was evaluated and compared to that of water rods. It was found that the only acceptable solid moderator is delta-phase zirconium hydride ($ZrH_{1.6}$), which generates a relatively high multiplication factor and a negative coolant void reactivity coefficient. Several issues key to the chemical and thermo-mechanical feasibility of $ZrH_{1.6}$ were assessed including zirconium-hydride/water interaction, hydrogen release, hydrogen redistribution, pressurization of the moderator box at high temperature, phase stability, and compatibility of zirconium hydride with the moderator box material. Zirconium hydride moderator rods appear to be suitable for use in SCWR thermal-spectrum cores, and therefore this approach will be pursued further during the project. Also, a simple analysis indicated that the use of zirconium-hydride moderator will not result in significant additional costs.

Task 2 - Fuel Cladding and Structural Material Corrosion and Stress Corrosion Cracking Studies: The design and fabrication of the University of Michigan supercritical water loop system for stress corrosion cracking tests was completed and experiments have begun (Figure 1 shows a view of the overall system). In this loop system, one tensile sample can be tested in various loading modes such as constant extension rate tension (CERT), constant load, ramp and hold, low cycle fatigue, and so forth. Additionally, six U-bend samples can be loaded into the test vessel, using sample holders secured to the vessel internal support plate. The initial test results indicate that type 304L stainless steel, which is commonly used in LWRs, is highly susceptible to stress corrosion cracking in SCWRs.

Figure 1. The photograph provides an overall view of the University of Michigan's supercritical water loop system.

The exposure facility at MIT incorporates a relatively large autoclave with an internal volume of approximately 860 ml. It is large enough to expose a rack of weight loss, welded, and U-bend samples for extended times. Initial experiments over a temperature range encompassing both subcritical and supercritical conditions have been completed with 316L stainless steel and Inconel-625 samples. Preliminary data from the Inconel-625 suggests the potential for localized breakdown and surface pitting both for exposed and occluded regions.

Task 3 - Plant Engineering and Reactor Safety Analysis: The preliminary core layout (dimensions, core configuration) and thermal-hydraulic design (temperature, pressure, flow rates) were developed. The design criteria for the system has been defined, and the correlations for heat transfer in supercritical water and the methods for the hot channel factors that will be used to verify the proposed design criteria for the fuel system have been identified. These criteria build on experience with LWRs, with one notable exception. Because critical heat-flux phenomena do not exist in SCWRs, it was decided that the key criterion for the SCWR core design would be the peak cladding temperature, analogous to the liquid-metal cooled fast reactor practice. Thus, a key step is having an accurate and reliable correlation for predicting the heat transfer coefficients in supercritical water cores. A review and assessment of the available correlations identified the Bishop and the Oka-Koshizuka correlations as appropriate and consistent. Three plant configurations (direct cycle, integral indirect cycle, and loop indirect cycle) have been selected for further investigation. An analysis was completed of the temperatures and density profiles in the average and hot channels for a variety of potential system configurations. Important parameters for this analysis were the core inlet and outlet temperatures and the option of canned assemblies versus an open lattice. A completely satisfactory design has not yet been identified. In preparation for the transient and accident analysis that will be conducted at INEEL in Years 2 and 3 of the project, the 3D finite-differences transient thermal-hydraulic/neutronic code, RELAP-ATHENA, was upgraded.

Planned Activities

Plans are to complete the work described above under research objectives during the next two years. The work during Year 1 was completed on schedule.

NUCLEAR ENERGY RESEARCH INITIATIVE

Particle-Bed, Gas-Cooled Fast Reactor (PB-GCFR) Design

Primary Investigator: Temitope A. Taiwo, Argonne National Laboratory

Collaborators: Brookhaven National Laboratory; Commissariat a l'Energie Atomique (CEA), France; University of Rome, Italy

Project Number: 01-022

Project Start Date: September 2001

Project End Date: September 2003

Research Objectives

The objective of this project is to develop a conceptual design of a particle-bed, gas-cooled fast reactor (PB-GCFR) core that meets the advanced reactor concept and enhanced proliferation-resistant goals of the U.S. Department of Energy's NERI program. The key innovation of this project is the application of a fast neutron spectrum environment to enhance both the passive safety and transmutation characteristics of the advanced particle-bed and pebble-bed reactor designs. The PB-GCFR design is expected to produce a high-efficiency system with a low unit cost. It is anticipated that the fast neutron spectrum would permit small-sized units (~150 MWe) that can be built quickly and packaged into modular units, and whose production can be readily expanded as the demand grows. Such a system could, therefore, be deployed globally. The goals of this two-year project are as follows:

(1) Design a reactor core that meets the future needs of the nuclear industry, by being passively safe with reduced need for engineered safety systems. This will entail an innovative core design incorporating new fuel form and type;

(2) Employ a proliferation-resistant fuel design and fuel cycle. This will be supported by a long-life core design that is refueled infrequently, and hence, reduces the potential for fuel diversion;

(3) Incorporate design features that permit use of the system as an efficient transmuter that could be employed for burning separated plutonium fuel or recycled LWR transuranic fuel, should the need arise; and

(4) Evaluate the fuel cycle for waste minimization and for the possibility of direct fuel disposal. The application of particle-bed fuel provides the promise of extremely high burnup and fission product protection barriers that may permit direct disposal.

Research Progress

Physics calculations have been performed in support of a reference compact fast-spectrum core, based on the pebble-bed design. The study investigated the impacts of the fuel-pebble packing fraction, and fuel material form and temperature on the potential for obtaining a sustainable critical core for a long-life design. Different fuel matrix and reflector materials were also investigated. The results have provided indications that mixed uranium and transuranics (TRU) carbide and nitride fuel forms are attractive for meeting the goals of a long-life core and high temperature operation (see Figure 1). Potential matrix materials for these fuel forms are titanium nitride (TiN) and zirconium carbide (ZrC), although enrichment considerations for the nitride fuel might make carbide fuel the preferred choice. The application of both enriched uranium and weapons grade constituents of the fuel has been discarded in the current planning because of the potential proliferation issues that could arise from their use and because they generally result in a higher reactivity swing than TRU fuel.

The physics studies also indicated that the goal of a long-life core (with a conversion ratio greater than one) requires a fuel volume fraction that is larger than the one currently used in high temperature reactor (HTR) designs. A high core fuel volume fraction (of ~30 percent) or core size (power density less than 25 W/cc) is required to get the desired sustainable design with a high conversion ratio. Achieving a high fuel volume fraction would require a redesign of the typical coated fuel particles, since it implies the minimization of the volume occupied by low-density carbon buffer and SiC zones to accommodate more fuel. Calculations made under this assumption, using (TRU,U)N in TiN-15 matrix or (TRU,U)C in ZrC matrix, indicated that a high conversion ratio (of greater than 1.2) can be obtained with low reactivity losses over the 15 to 30 years of irradiation.

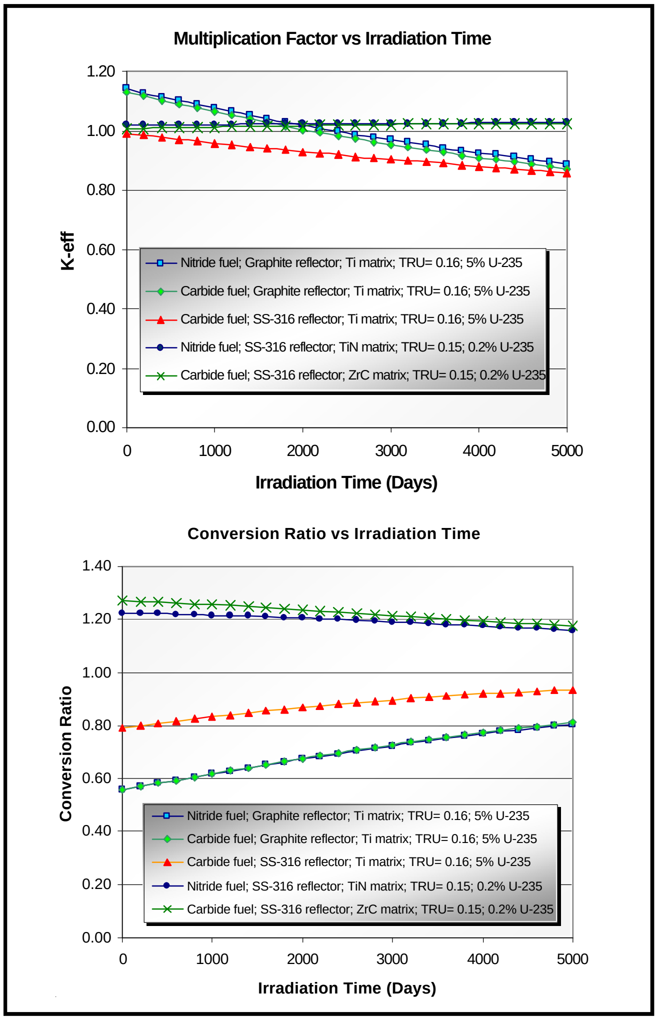


Figure 1. The graphs present the multiplication factor and conversion ratio as a function of irradiation time for PB-GCFR designs.

It is known that the safety case for fast spectrum reactors employing gas-cooling is complicated by the poor heat transfer properties and low thermal inertia of the gas coolant. The ability of this reactor type to survive a scrammed depressurization accident with a concurrent loss of electrical power, without undue hazard to the public, is clearly an attractive feature of an advanced GCFR of the Generation IV class. Researchers associated with this project have explored, and continue to examine, a number of concepts that could potentially provide this safety feature through passive means. A fundamental assessment was first performed of heat transfer modes and the implications of the decay heat curve. Scoping thermal calculations were carried out. The study revealed that natural convection at 1 atmosphere cannot be relied upon for the available selection of primary coolant gases, and that radiation through the coolant would dominate above 1,000°C. Below this temperature, a better alternative may be to provide conduction pathways. For the period immediately following scram, however, heat transfer on this timescale is not adequate for the core materials of the foreseeable future, and substantial core thermal inertia is necessary. The results also indicated that to improve the feasibility of the passive core concept, it would be prudent to reduce the reactor power envelope to below 300 MWt. Three types of basic core elements were investigated with the potential to provide core configurations that have the desired passive core safety feature: (i) block/plate, (ii) pin/tube, and (iii) pebble/particle. After the initial investigation, it was decided that the major focus of the work would be on the pebble/particle fuel element and, in particular, on the pebble-bed core configuration.

A unique concept was introduced to increase the heat storage capacity of the fuel pebble. This concept uses fuel spheres in which the center is filled with a material that does not contain fuel and which can melt and absorb heat as latent heat of fusion. This concept also substantially lowered the temperature rise within each fueled pebble. The scoping studies and the new pebble concept led to a conceptual design for an annular pebble-bed fast reactor. A severe depressurization accident was simulated for this design choice with a two-dimensional conduction model. The analysis predicted a maximum fuel temperature of 1,627°C, which is only 27°C above the current limit that is arbitrarily based on the limit for graphite fuel. The reactor vessel temperature was observed to drop sharply at the initiation of the transient, which includes a scram, and to subsequently recover to a local maximum before the final decline, which lasted until the end of the accident. The local maximum temperature of 489°C is safely below the assumed allowed value of 537°C. However, the active core power density is limited to 23 W/cc. Since this may not be economical for a fast reactor, three additional concepts were developed: (1) prompt unloading of the pebble fuel, (2) extended flow coastdown, and (3) tube reactor with a tank.

For prompt unloading of pebble fuel, during a severe depressurization accident, all of the fuel pebbles in a pebble bed reactor would be quickly dropped into a series of storage tanks. The tanks would be located inside a borated water bath. The decay heat would be conducted through the tank walls and would boil the water. The steam would rise through pipes to an air-cooled steam condenser located outside of the reactor building. A tall cooling tower over the condenser would provide a natural draft that would drive the air flow needed to cool the condenser. If the system capacity is too small to handle the entire decay heat load during the earlier part of the accident, some of the steam could be released to the atmosphere via a pressure-relief valve. The results indicate that the removal of several megawatts on a steady-state basis is quite possible with a cooling tower that is 70 feet tall and 25 feet in inside diameter at the throat.

In practice, the extended flow coastdown option is almost an instantaneous event: the reactor scram reduces the power level from full operating power down to decay levels in a fraction of a second. However, it takes many seconds for the system to depressurize and many seconds for the blowers to coast down and stop. This "coastdown" could also be extended through passive mechanisms. The ignored convective flow rate in the initial portion of the accident could potentially remove a substantial amount of decay energy and cause the predicted peak temperatures to be much lower than those predicted by the earlier analysis. With this in mind, a dynamic model was developed to include realistic pressure histories and flow coastdowns in the analysis. Results obtained showed that the results are marginal. In the case of the tube/tank reactor design, a small (long and thin) spaghetti core is proposed, composed of approximately 4 tubes (each, 0.3 meters in diameter x 4 meters in length) arranged in an array around the control rods. The tubes could be filled with fuel pebbles and would be internally cooled with high-pressure helium (~7 MPa) while the tank (calandria) surrounding the tubes would contain low pressure carbon dioxide (1 MPa) to remove decay heat at natural convection conditions. Initial calculations show that a chimney height of approximately 12 meters is required to remove approximately 1 percent decay heat.

Pursuing further the concept of combining passive conduction/radiation heat transfer with natural convection, the cold finger concept for total passive decay heat removal was conceived and developed. Cold fingers provide a passive means of decay heat removal during severe depressurization accidents. These fingers are bayonet heat exchangers, which also house the control rods and thereby serve dual purpose as both reactivity control devices and a passive decay heat removal mechanism. These fingers are inserted parallel to the axis of the core and are evenly distributed throughout its circular cross sectional area. A parametric study was performed leading to three candidate core designs that best meet steady-state thermal and hydraulic requirements for a 300 MWt gas-cooled pebble-bed reactor operating at rated conditions. Safety analyses to investigate the performance of the cold finger concept were then carried out with these core designs. The results are promising. The cold finger concept will be further explored in the second year work through the implementation of dynamic models, and dynamic analyses will be performed.

A literature review was performed to evaluate candidate fuel and materials compatibilities, high-temperature mechanical and thermal properties, and performance issues expected by operation in a fast neutron spectrum (E>0.1 MeV). Much of the effort was devoted to identifying sources of pertinent information, collecting material properties, and reviewing current gas-cooled reactor fuel designs. Space-reactor developmental efforts conducted in the 1960s were also evaluated. The literature review was completed to identify candidate fuels and materials for the development of a new GCFR to meet Generation-IV criteria.

In a similar manner, property data for industrially available structural materials with well-established manufacturing technologies were collected and assessed to identify candidate materials for various key components of the PB-GCFR. Since detailed design information about these components do not presently exist, materials and materials systems that were evaluated in other reactor-development projects were considered first. Based on this evaluation, recommendations for structural materials have been made for structures in the vicinity of the fuel zone (including ceramics such as SiC, ZrC, TiC, MgO, $Zr(Y)O_2$, TiN, and Si_3N_4); for the pressure vessel (2¼ Cr-1Mo and 9-12Cr steel); cooling system components (Inconel 718, Inconel 800, and Hastelloy X); shielding and thermal barriers (borated Type 304 and 316 stainless steel, ferritic HT9,and various vanadium alloys); and the reflector zone (uranium, tungsten, iron, stainless steel, and Nb-1Zr).

Planned Activities

Physics issues that have not been investigated in detail for the long-life, PB-GCFR core will be pursued in the future. These include determining whether a single-batch fuel management scheme would require an annular core design, particularly at large core volumes, and whether enrichment splitting is required. Additionally, it is necessary to calculate core reactivity coefficients and employ them in safety analysis. Besides long-life core designs, one can envision other designs that enhance the proliferation resistance of nuclear power systems. One approach is a core design permitting the efficient and deep burning (very high burnup) of the fuel material. By using the coated-particle fuel form in the fast neutron spectrum environment, it could be possible to effectively burn the fuel. The fast spectrum is particularly suited for this task, since all TRU nuclides can be fissioned in this energy range. An evaluation of the feasibility of using the GCFR as an efficient TRU burner that enhances proliferation resistance is being planned for this project.

A detailed core design based on the cold finger concept is planned for the next phase of the project (Year 2). Heat transfer calculations for the depressurization accident with concurrent loss of electric power will be performed for this design. Additionally, when a specific core design configuration and fuel form has been selected, scoping calculations will be performed to assess the flow induced vibration and thermal stress potential of the design.

NUCLEAR ENERGY RESEARCH INITIATIVE

A Miniature Scintillation-Based, In-Core, Self-Powered Flux and Temperature Probe for HTGRs

Primary Investigator: David E. Holcomb, Oak Ridge National Laboratory

Collaborators: The Ohio State University

Project Number: 01-039

Project Start Date: August 2001

Project End Date: September 2004

Research Objectives

The objective of the proposed project is to develop a miniature scintillation-based, in-core, self-powered neutron flux and temperature probe. The probe would be generally applicable to any reactor technology, but would be specifically designed for the temperatures of high temperature gas reactors (HTGRs). The scintillation assembly consists of a uranium layer placed against a thick film scintillator layer. The fission fragments resulting from neutron interactions in the uranium produce light in the scintillator. The light from the scintillator is guided out from the core using a hollow core optical fiber. Both the converter layer and the scintillator are segmented. A scintillator of one wavelength is collocated with a lightly enriched ^{235}U uranium layer. A scintillator with a different characteristic wavelength is placed against a higher enrichment ^{235}U uranium layer (2 percent and 4 percent, for example). The scintillators produce different wavelengths of light, allowing independent readout of the scintillation. A conceptual layout of the probe tip is displayed as Figure 1.

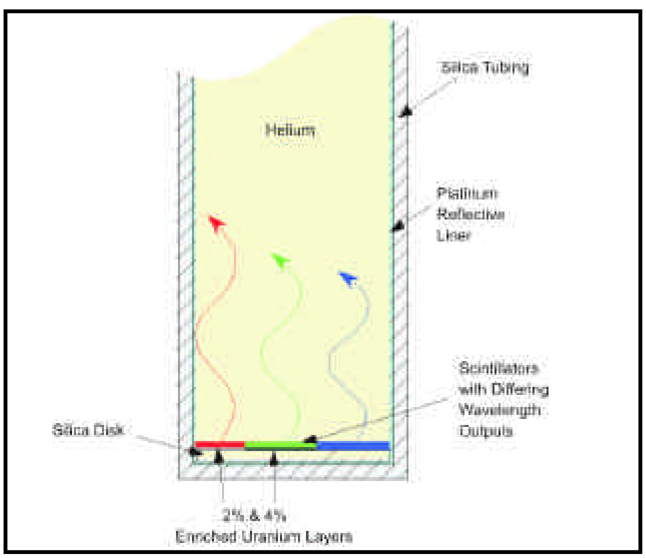


Figure 1. The graphic illustrates the conceptual layout for the scintillation probe tip.

At low temperatures (below ~400 °C), neutron flux is indicated by a weighted sum of the number of scintillation pulses produced by each scintillator (pulse mode). At higher temperatures, neutron flux is indicated by the total amount of light produced by either scintillator with the ratio of the different amount of light at the different waveband intensities serving as a burn-up monitor (current mode).

The probe indicates temperature in two manners: at higher temperatures multi-wavelength pyrometry is employed, while at lower temperatures this is supplemented by observing the optical decay time variance of the scintillation pulses with temperature. Total pulse count history is used to compensate for burn-up of the converter atoms. The detector is anticipated to have minimal gamma pulse response due to the thinness of the scintillation layer. However, the gamma response that does exist is compensated for using a scintillator layer without the neutron sensitive uranium undercoating. The thinness of the scintillator layer also minimizes the effects of the increased self-absorption (radiation darkening) with use of the scintillator material.

Research Progress

The project began with an initial survey of the types of scintillators likely to be useful as well as phosphor layer deposition technologies. As a result of the survey, a scintillation layer deposition technique closely analogous to that employed in the manufacture of cathode ray tubes was selected and refined to the purposes of the project. Essentially, the phosphor powder is suspended in a chemically compatible liquid along with an inorganic binder. The mixture is then centrifuged down onto a substrate, the suspension liquid is decanted off, and the binder material is cured to affix the scintillator to the substrate. This is a general phosphor powder deposition process, providing the capability to deposit almost arbitrary types of scintillators.

Simultaneously with developing the capability to deposit phosphor spots, a laboratory apparatus was designed and fabricated to allow measurement of scintillator characteristics as a function of temperature under alpha particle irradiation. The central portion of the apparatus is contained within a vacuum chamber. A computer-controlled heater is mounted within the vacuum chamber and scintillator samples are placed on the heater surface. A collimated alpha source is pointed at the sample to produce scintillation pulses. The scintillation light is guided out of the chamber using a two-element lens system coupled to a chilled photo multiplier tube (PMT) with near single photon counting capability. The light is filtered before arriving at the PMT with a short wavelength-passing filter to remove as much as possible of the blackbody light emitted by the hot sample.

It has become apparent as scintillator materials were tested at progressively higher temperatures that blackbody emissions become the dominant source of light as compared to individual neutron induced pulses above approximately 400 °C. While this is advantageous for measuring temperature via multi-wavelength pyrometry, the intensity of blackbody emission in the visible bands at higher temperatures does mean that it will not be possible to operate the detector in pulse mode at higher temperatures. This is not believed to be particularly significant since current mode is available for high-flux, high-temperature situations. Figure 2 shows the developed scintillator testing apparatus in both a photograph and a schematic illustration.

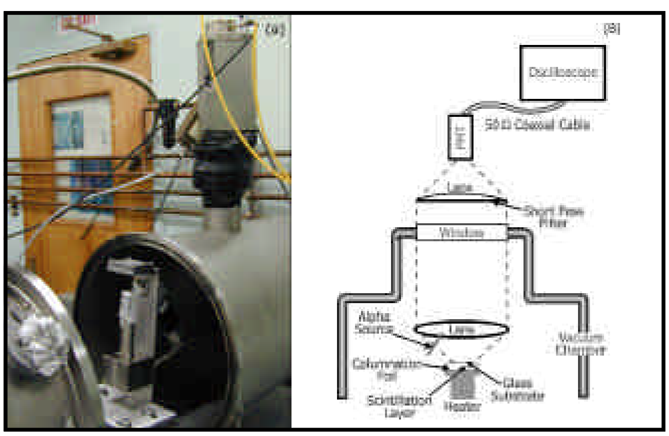


Figure 2. The laboratory scintillator testing apparatus appears both photographically (a) and schematically (b).

Significant progress has been made towards scintillation probe mechanical fabrication. Additionally, an aqueous technique to deposit silver films onto the interior surface of the light guides has been implemented, as has a slurry-based polishing technique to obtain a mirror finish in the light guides. While this technique is only applicable for light guides not exposed to temperatures above approximately 900°C due to silver vaporization, the technique is simple and requires no expensive equipment thus these light guides may be preferred in certain situations.

Planned Activities

A platinum chemical vapor deposition process has been developed for higher temperature applications (limited by the fused silica maximum use temperatures of approximately 1,100°C). Thus far the process the process has only been demonstrated on short segments and researchers anticipate scaling the process as well as performing confirmatory survivability testing over the next two quarters.

Ohio State has made significant progress towards implementing the high temperature, near-core reactor testing environment and apparatus. Over the course of the next quarter, it is anticipated that nuclear testing will begin at OSU. The testing will include both lower flux and temperature testing in a subcritical pile; gamma only response testing using OSU's ^{60}Co irradiator; reactor testing at ambient temperature; and finally, elevated temperature reactor testing.

NUCLEAR ENERGY RESEARCH INITIATIVE

Construction Cost Reductions in Generation IV Nuclear Energy System using Virtual Environments

Primary Investigator: Timothy Shaw, Pennsylvania State University

Collaborators: Panlyon Technologies; Westinghouse Electric Company; Burns and Roe Enterprises

Project Number: 01-069

Project Start Date: August 2001

Project End Date: September 2004

Research Objectives

The objective of this project is to demonstrate the feasibility and effectiveness of using full-scale virtual reality simulation in the design of future nuclear power plants. Specifically, this project will test the suitability of Immersive Projection Display (IPD) technology to allow engineers to evaluate the potential cost reductions that can be realized in installation and construction sequences for Generation IV Nuclear Energy Systems. The intent is to see if this type of information technology can be used to improve arrangements and reduce both construction and maintenance costs, as has been done by building full-scale physical mock-ups for other systems.

Research Progress

The contract was awarded in August 2001 and work began shortly thereafter. During a kickoff meeting, participants were introduced to the IPD technology. Potential uses as well as benefits and shortcomings of the technology were discussed. A number of test models were converted from existing virtual reality modeling language (VRML) files, downloaded from the Internet. This demonstration allowed half of the file conversion process to be tested. The test was expanded to include a number of different file formats available from the 3-D computer-aided design (CAD) package that Westinghouse used in the design of the AP600 plant. A method was developed for transferring and converting CAD files to a format that could be displayed in the IPD.

Once a suitable file conversion process was found, work began on developing the functionality of the software used to manipulate and interact with the virtual environment. Many features were added to the software over the course of the year. Several examples follow.

- A virtual measuring tape allows the user to quickly determine distances and clearances between objects.
- A virtual crane allows objects within the space to be moved in a manner similar to an actual crane.
- Components could be resolved separately and then tagged for identification, which allows the user to determine to which fluid system a piece of equipment belongs.
- Some of the valves have been modeled individually to allow them to be grabbed and moved or potentially operated.
- A system of hand gestures and voice-activated commands allows the user to point at components and identify them, toggle a measuring tape, operate a crane, or grab and move objects. Some of these interactions are depicted in Figure 1.

A simple collision detection scheme, which notifies the user if objects are touching, has been investigated for use in the installation sequence study and the maintenance evolution. Interaction with the environment is still somewhat awkward, but the software is being continually updated and refined. A configuration file allows the user to pre-program viewpoints and set the navigation speed.

Room 12306 within the Westinghouse Advanced Passive (AP 600/1000) Plant was chosen as a test bed for the virtual mock-up due to its relative complexity. The room is designed to contain five modules and piping assemblies, which are composed of piping from ten different fluid systems. The original files received from Westinghouse contained all of the geometry in a single VRML 1.0 file. The next iteration broke the room down by fluid system. The latest iteration has a large number of the valves modeled individually so they can be selected and manipulated. Other components are resolved by system. Methods for taking the existing geometry and increasing the resolution have been investigated so that

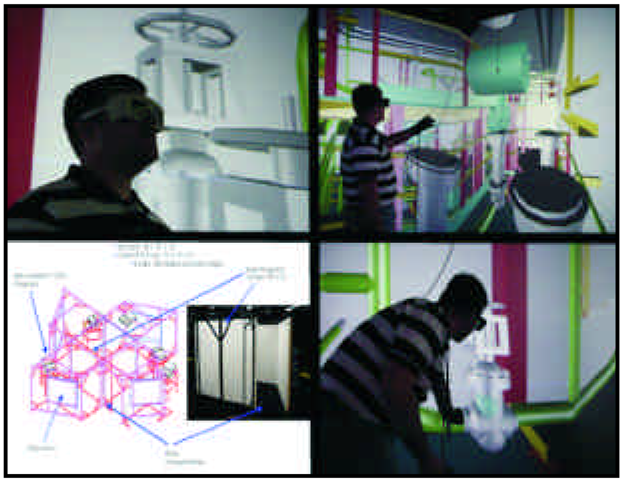


Figure 1: The graphics show features of the interactions possible using the virtual reality software being used to enhance construction studies for new nuclear power plants: (clockwise from upper left) Active Stereo Glasses, Virtual Crane, Interaction with Environment, and IPD Layout.

the individual pieces of piping that connect modules, called make-up or spool pieces, may be modeled.

During the initial development of the virtual mock-up, the room was reviewed by construction personnel to determine whether or not the technology provided sufficient realism, thereby providing a potential design tool. A survey was completed by the collaborators, which compared the virtual mock-up to computer models (3-D CAD) and physical mock-ups. Results of the surveys are currently being compiled.

In addition to the enhancement of room 12306, a lower fidelity virtual mock-up of the containment of the AP600 was constructed for educational and demonstration purposes. This model has been shown to industry personnel and government officials at the ICONE-10 and G-8 Energy Ministers' meetings. Response to the technology's potential has been very positive.

A number of tools have been developed to support the study of the installation sequence of the developed virtual mockup. Currently, the user is able to watch the 4D installation of prefabricated modules and makeup pieces. Different colored models are used to show the components before, during, and after installation. The software allows parallel activities to be displayed, as well. Installation sequences may be played back or viewed un-timed step-by-step within the virtual reality environment. Tools that assist the user in developing an installation sequence have been developed. A software "SELECT" function allows the user to change the color of models as they are installed in the space. Upon completion, the entire room appears in red. This information can be quickly loaded into the system, and the installation sequence can be played back for review.

Planned Activities

Two installation sequence studies are being developed. The first study asks groups of construction management students to attempt to develop a construction sequence after a brief introduction to the components in the room. The students are given the freedom to change module boundaries and to cut pipes. The sequence they develop is then played back and evaluated. The second study asks experienced construction superintendents to develop a schedule using the isometric drawings provided by the designer. The schedule they develop will then be loaded into the virtual environment system for review and evaluation to identify constructability issues not revealed using the isometric drawings. The subjects will be surveyed concerning the benefits or drawbacks of developing the installation sequence with the assistance of the virtual environment technology.

NUCLEAR ENERGY RESEARCH INITIATIVE

On-Line NDE for Advanced Reactor Designs

Primary Investigator: Norio Nakagawa, Ames Laboratory

Collaborators: Westinghouse Electric Company, LLC

Project Number: 01-076

Project Start Date: October 2001

Project End Date: September 2004

Research Objectives

The extended refueling interval of Generation IV nuclear reactors creates new maintenance challenges. Current commercial reactors achieve high levels of availability and reliability by employing outage-based maintenance (i.e., performing methodical, periodic, off-line inspections, preventive maintenance, and component repair/replacement during planned refueling periods). Compared to the traditional 1- to 1.5-year refueling cycle, Generation IV reactors use extended refueling intervals such as 4 years and beyond. New approaches are required to keep maintenance from interrupting operation. The key strategy of this effort is to replace/augment current outage-based maintenance by on-line structural health monitoring, to ensure the current level of safety.

The project's specific objectives follow:

(1) Determine, based upon regulatory requirements and commercial reactor experience, the most appropriate inspection through a comprehensive review of each component of the Generation IV reactor design.

(2) Determine which inspections provide the greatest economic benefit when implemented as in-situ monitoring, in light of the inspection requirements and reduced off-line inspection opportunities.

(3) Optimize mechanical design parameters to simplify inspection.

(4) Develop the concept of a built-in structural integrity monitoring system using electromagnetic, ultrasonic, or radiation detectors, to be integrated into design for Generation IV nuclear power systems.

(5) Evaluate and characterize the performance of conceptual sensor systems by the use of physics-based simulation models.

(6) Enhance the capabilities of the simulation models to meet the challenges posed by unique power system environments.

(7) Select sensor types and materials, find their compatibility with hazardous environments, and examine their possible degradation.

Among various Generation IV reactor designs, the team pays particular attention to the International Reactor Innovative and Secure (IRIS) design. As with other Generation IV reactors, it has several general design goals: a) passive safety features; b) increased availability and economy; c) long-term, uninterrupted operation; d) environmental friendliness; and e) proliferation resistance. To meet these design requirements, most Generation IV reactors use compact, integrated designs, and are made compatible to operating with extended refueling cycles. IRIS, in particular, integrates the reactor core, steam generators (SG), and primary-coolant pumps in a single reactor pressure vessel (RPV). By design, the IRIS reactor is compact and cost-effective, and requires less maintenance because it has no large primary-water loop piping outside the RPV. In addition, there is less likelihood that the SG tubes will develop stress-corrosion cracking (SSC) since they operate in compression. However, the long refueling cycle (every four years for the IRIS-based design) poses a maintenance challenge because there will be fewer opportunities for periodic outage-based maintenance, as has been practiced for existing commercial reactors.

From the point of view of these key design features, future reactors such as IRIS call for a new maintenance strategy, i.e., not only relying on outage-based maintenance, but also actively performing on-line inspection and monitoring. The advantages of on-line health monitoring are multi-fold:

1) It is done while the reactor is in operation, not requiring shutdown for inspection/monitoring activities.

2) It is done remotely, thus greatly reducing exposure levels.

3) It allows investigators to perform continuous or on-demand system integrity verification, which means that deviation from normal operation can be detected in real time, maximizing potential options for issue resolution.

To demonstrate the concept, the project develops conceptual on-line sensor systems that will replace/augment outage-based maintenance.

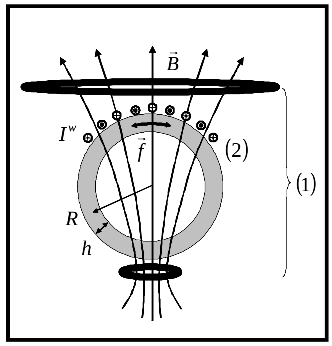


Figure 2. In this schematic of an EMAT system, the asymmetrical Helmholtz coil pair (1) will generate DC fields approximately perpendicular to the tube wall, suitable to generate torsional modes.

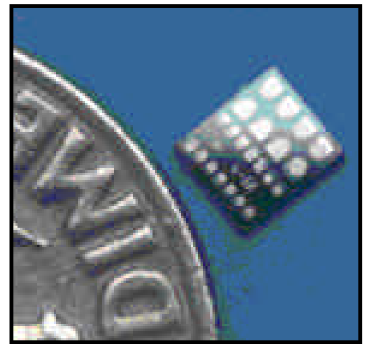


Figure 3. The photograph shows a patented radiation detector fabricated on a SiC semiconductor chip.specific design uses an encircling solenoid coil surrounding several tubes chosen by sampling.

Research Progress

First, the IRIS design was reviewed to identify critical inspection needs. To date, several candidates for on-line monitoring have been identified: steam generators (SG), the reactor pressure vessel (RPV), reactor core, and coolant pumps. The potential issues identified for steam generators are magnetite deposit buildup, tube mount integrity, and tube integrity, while the concerns for the RPV and core are cracking and fuel activity anomaly, respectively. The coolant pumps may cause fatigue in their mounting mechanisms, while failure of a pump will create coolant flow anomaly inside RPV, leading to reactor instabilities.

Second, the team conceived on-line monitoring systems that can address identified monitoring needs, based on several candidate NDE methodologies. Specifically for SG, likely candidates are the eddy current (EC) method via fixed-site coils (Figure 1) and the ultrasonic guided-wave method via electromagnetic-acoustic transducers (EMAT) (Figure 2). For the RPV cracking and core anomaly monitoring, the uses of ultrasound testing (UT) via EMATs and radiation-based techniques based on silicon-carbide (SiC) detectors (Figure 3), respectively, will be explored. More recently, the above-mentioned pump issues were recognized, and the team has begun to explore the use of the use of hard-ray emission prompted by radioactive nitrogen-16 isotopes contained in the primary coolant.

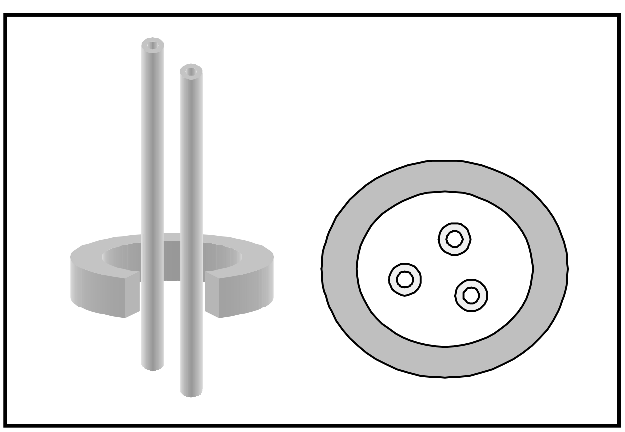


Figure 1. The diagram shows the conceptual encircling-coil design of an on-line EC inspection for magnetite deposit detection. A specific design uses an encircling solenoid coil surrounding several tubes chosen by sampling.

The third part of the team's activities has involved model-based studies of conceived on-line monitoring sensors to estimate their performance. In particular, EC sensor signals were estimated based on a simplified model. It was found that when the EC coils are operated at 20-30 kHz, they will produce distinctive magnetite-deposit signals, clearly standing out of the geometry-signal background. Guided-wave modes traveling along a tube have been worked out for the EMAT UT application. Among these, torsional modes are the primary candidates for use. A conceptual sensor system design was developed (Figure 2), in which the DC bias magnetic field distribution was actually computed to confirm the field lines drawn intuitively in Figure 2. A radiation propagation model based on transport equations is being developed for studying radiation-based techniques. The model involves both photon and charged-particle (electron and positron) sectors. The conceptual design of the algorithm has been completed.

Planned Activities

These foregoing results and outputs indicate that the project is on schedule. Since no particular issues/concerns have been identified, the team plans to follow the original project schedule. The Year Two goals follow:

(1) Determine, by prediction, neutron and gamma-ray fluxes in critical areas under normal reactor operation.

(2) Complete integration of the charged-particle transport code into the photon transport code.

(3) Select a likely set of candidate on-line electromagnetic sensor designs according to the requirements of on-line SG tubing inspections.

(4) Complete the upgrade of EC model code for array probes.

(5) Determine the beam model for each of the EMAT designs.

In addition to these primary activities, the team will continue to interact with the IRIS design team members by participating in their team meetings, in order to mutually disseminate project developments. Exploration of additional on-line monitoring needs and methodologies will also be continued as further design developments of IRIS, and possibly of other Generation IV designs, become available.

Finally, public dissemination will continue of the team's on-line health-monitoring concept and its benefits when applied to nuclear power systems. The target audience includes broad technical communities, especially the NDE community.

NUCLEAR ENERGY RESEARCH INITIATIVE

Supercritical Water Nuclear Steam Supply System: Innovations in Materials, Neutronics, and Thermal-Hydraulics

Primary Investigator: M. Corradini, University of Wisconsin-Madison (UW)

Collaborators: Argonne National Laboratory (ANL)

Project Number: 01-091

Project Start Date: August 2001

Project End Date: September 2004

Research Objectives

A nuclear reactor cooled by supercritical water[1] with a once-through direct power cycle would have a much higher thermal efficiency than a standard water reactor, and could be based on standardized water reactor components (light water or heavy water). The theoretical efficiency could be improved by more than 33 percent over that of other water reactors and could be simplified with higher reliability (e.g., a boiling water reactor without steam separators or dryers). Such improvements would be accompanied by a corresponding decrease in the nuclear plant levelized electricity cost, and thus, could make this nuclear steam supply system quite competitive in future electric power markets as a centralized power source. In addition, this concept would take full advantage of 50 years of light water reactor (LWR) technology, would allow for incremental as well as substantial improvements in reactor technology to maintain and enhance safety and reliability, and would provide flexibility in the fuel cycle to allow for substantial improvements in sustainability. This research project will make such a system technologically feasible by accomplishing the following objectives:

(1) Employ innovative ion implantation surface modification techniques to improve material compatibility at supercritical conditions. Plasma Source Ion Implantation techniques will be used to modify clad materials and demonstrate improved corrosion/wear resistance under supercritical thermal-hydraulic conditions.

(2) Use neutronics analyses to identify ranges of alternative fuel cycles, including variations in enrichment, refueling schedules, recycling, and conversion/breeding. These analyses would focus on coolant density effects at supercritical conditions to verify passive safety with comparisons from the standpoint of fuel burnup, flexibility, proliferation resistance as well as sustainable development, using quantitative metrics.

(3) Conduct thermal-hydraulic studies that focus on heat transfer and flow stability issues associated with coolant density changes for natural circulation of supercritical water. Scaled simulation experiments are to be designed and performed to provide heat transfer and stability data to be used in developing predictive tools.

[1] T_c=375°, P_c=220 ATM, Vapor Density and Liquid Density converge

Research Progress

Work has gone forward on three tasks.

Task 1. Cladding Materials, Surface Treatment, and Corrosion Loop: Three cladding materials—Inconel 718, A1SI 316 austenitic stainless steel, and Zircaloy-2—were selected for this task. Samples of these materials have been obtained and surface preparation and coating were completed on several of the samples. Table 1 summarizes the three fundamentally different approaches of plasma surface modification that is underway.

Table 1. Various Plasma-Based Surface Treatments

Plasma surface treatment	Improvements anticipated	Possible mechanisms
Ion implantation - Room temperature* - Elevated temperature	Hardness, wear, corrosion & oxidation resistance	Formation of hard compound phases, alteration of oxide nucleation mechanism and deeper modified layer (for elevated temperature)
Non-equilibrium surface alloying	Oxidation resistance	Formation of a more adherent oxide with slower growth kinetics
Heavy ion bombardment	Corrosion and oxidation resistance	Surface microstructural homogenization due to energetic ion-induced mixing

*Underway this quarter

A supercritical water flow loop for corrosion studies has been designed and engineering work for construction has begun. Figure 1 shows the loop design along with the high heat flux heater design, which utilizes molten lead as the heat transfer media. The loop has an inner tube diameter of 4.29 cm and an outer tube diameter of 6.03 cm (Inconel 625).

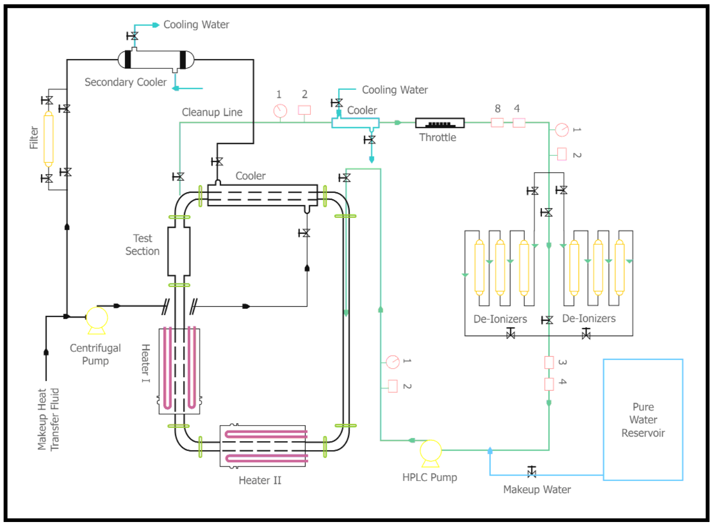


Figure 1. Flow Loop for SCW System

Task 2. Neutronic Analysis of Coolant Density Effects and Cladding Materials: The neutron physics of a supercritical water reactor (SCWR) is different from that of a standard (LWR) because of the low water density of supercritical water, high operating temperature of the SCWR, and the different cladding materials. In particular, the low density of supercritical water near the pseudo-critical temperature results in a harder spectrum in the SCWR related to a standard LWR. Although the WIMS8 code is well-developed for the LWRs the validation of the WIMS8 code is necessary because the neutron spectrum of the SCWR is different from that of the standard LWR. For the first stage of validation of WIMS8 code, the MCNP4C was used for benchmarking purposes.

The k_∞ value and normalized pin power distributions of the 17x17 Westinghouse standard assembly, operating with a coolant/moderator density of 0.3 g/cm^3, predicted by MCNP4C and WIMS8, are provided in Table 2. The WIMS8 power distributions presented in this table were calculated with a 28-group transport solution. The maximum observed differences in k_∞ are 387 pcm (0.3 percent). The normalized pin powers of the 28-group WIMS8 calculation are generally within c1c of the MCNP4C result and the root mean squares (RMS) of the power differences between WIMS8a and MCNP4C are very similar to the RMS of the statistical error of the MCNP4C.

Table 2. Summary of the MCNP4C and WIMS8 Calculations

Calculation conditions	MCNP4C		WIMS8		
	k_∞	RMS of MCNP power error a)	Number of group b)	Difference of k_∞, pcm c)	RMS of power error d)
U enrichment=5.0% Water density=0.3g/cm^3 Clad material=Zr-2 Temperature=300 K	1.27325 c 0.00028	c 0.44	6	-211	0.47
			28	-387	0.40
			172	-318	0.40

a) Root mean square of the MCNP relative power error
b) Number of neutron energy groups in the transport solution
c) Difference of the $k_\infty = 10^5 * (k_\infty^{WIMS} - k_\infty^{MCNP})$, pcm
d) Root mean square of the difference power error between MCNP and WIMS8a

Task 3. Natural Circulation Heat Transfer and Flow Stability Studies: The design of a natural circulation loop of supercritical carbon dioxide is being finalized. The purpose of this test loop is to investigate natural circulation heat transfer and flow instabilities associated with the use of a supercritical fluid. Such instabilities are of interest because a number of proposed nuclear reactor systems with supercritical water would utilize the large density change across the pseudo-critical point to drive natural circulation cooling. Flow instability in the circulation may adversely impact the ability of the system to remove heat from the reactor core.

The design of the loop is shown schematically in Figure 2. It is a rectangular loop that consists of horizontal heating and cooling sections connected by two vertical pipes. The inner diameter and height of the loop are 0.014 meters and 2.0 meters, respectively. Calculations were performed for the loop operating at a constant pressure of 80 bar and a constant inlet temperature of 28°C. One of the operating curves produced by the calculations was the plot of the steady-state natural convection mass flow rate vs. the heater power. The plot showed that a peak in the mass flow rate occurs around the power needed to take the fluid through the pseudo-critical temperature along the heated section. Chatoorgoon recently proposed an ad hoc criterion postulating that this peak would correspond to an instability boundary [Int. J. Heat Mass Transfer, 44 (2001)]. This criterion suggests that a new type of flow instability, different from traditional instabilities, would arise in the region of negative slope on the plot of mass velocity vs. power. The natural circulation loop has been designed to investigate (and confirm or deny) this criterion. Specifications for the loop heating and cooling power as well as the dimensions (height and diameter) are consistent with this design requirement.

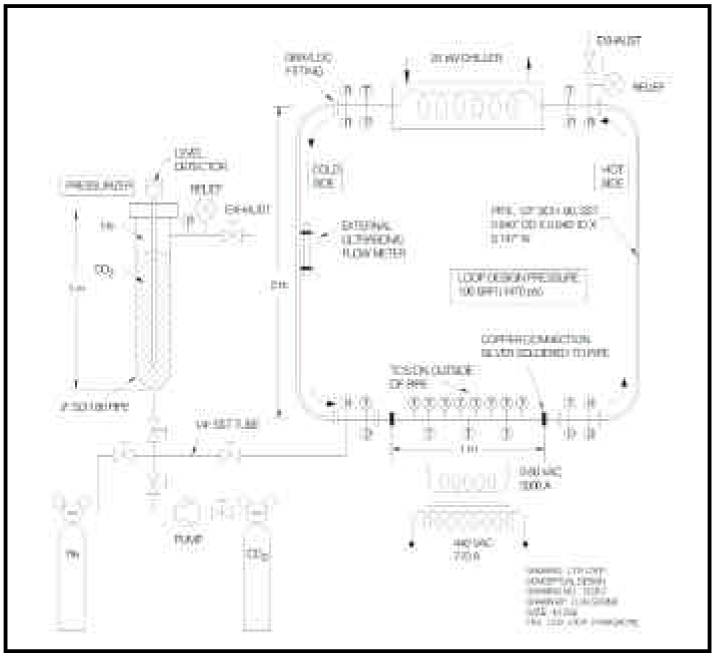


Figure 2. Supercritical CO_2 Natural Circulation Loop Design

Planned Activities

Three future tasks have been planned.

Task 1. Initial tests of the surface-treated samples will be conducted to obtain a baseline analysis of the modified surface. These baseline tests will include potentiodynamic corrosion testing, wear, synergistic effects of wear and corrosion, and electron microscopy of the three reactor alloy materials being investigated in this project, before and after plasma surface treatments.

The supercritical water loop for corrosion studies will be constructed and become operational by the end of this calendar year. Once the loop is operational, the surface-treated samples will be tested in a prototypical supercritical water environment, and the test results will be compared to those of the baseline tests.

Task 2. The good agreement of WIMS8 with the MCNP4C calculations indicates that WIMS8 code is well-suited for predicting the eigenvalues and power distribution under the SCWR environment. However, since this benchmark has been done at the room temperature, the temperature effects on the cross-sections were not taken into account in this benchmark. Therefore, additional benchmark calculations will be performed after preparing the cross sections at the operating temperature.

Task 3. The supercritical CO_2 loop for studies of natural circulation heat transfer and flow stability will be constructed and become operational by the end of this calendar year. An experimental plan for proposed tests will be developed and issued. A safety plan for the experiments will be prepared to comply with the Argonne National Laboratory's environmental safety and health requirements.

NUCLEAR ENERGY RESEARCH INITIATIVE

Testing of Passive Safety System Performance for Higher Power Advanced Reactors

Primary Investigator: Jose N. Reyes, Oregon State University (OSU)

Project Number: 01-094

Project Start Date: August 2001

Project End Date: September 2004

Research Objectives

The objective of this project is to assess the performance of various passive safety systems for advanced reactors operating at powers on the order of 3,000 MWth. The advantage of passive safety systems for core cooling following loss-of-coolant accidents is that they do not rely on safety grade pumps or alternating current power. Rather, they depend on natural driving forces such as gravity, compressed gas, and natural circulation to provide core cooling for an indefinite period of time after an accident. Currently, the AP600 is the only passively safe nuclear plant in the world that has received Design Certification from the U.S. Nuclear Regulatory Commission. Westinghouse has recently proposed the development of an AP1000 that would offer significant economic advantages over the AP600. Because of its multiple passive safety systems, the availability of a geometrically similar integral test facility at OSU, and the lower power AP600 database that can be used for purposes of comparison, the higher power AP1000 is an ideal candidate for this high power passive safety system study. The work will be carried out over a three-year period and includes a test facility scaling analysis, advanced plant experiment (APEX) facility modifications, and passive safety system tests and assessment for design basis loss-of-coolant accidents (LOCAs).

Research Progress

Figure 1 is a flow chart describing the overall OSU research program. Significant progress was made during the first two quarters of this project. The first activity to be completed was a scaling assessment of the geometric, kinematic, and dynamic similarity between the APEX test facility components and operating conditions and those of the AP1000 reactor. As shown in Figure 2, APEX is a 1/4-length scale, integral system, test facility originally designed to simulate the important thermal-hydraulic behavior of the Westinghouse AP600. The scaling analysis for this project has identified the modifications that are needed to adequately simulate the new AP1000 reactor design concept using the existing OSU APEX test facility. The APEX facility modifications will be as follows:

- Increase core power from 600 kW to 1.0 MWth
- Replace the Pressurizer and Surge line
- Enlarge the Core Make-up Tanks (CMT)
- Enlarge Automatic Depressurization System (ADS) 4 valve flow area
- Reduce the CMT and Incontainment Refueling Water Storage Tank (IRWST) line resistances
- Upgrade the Data Acquisition System

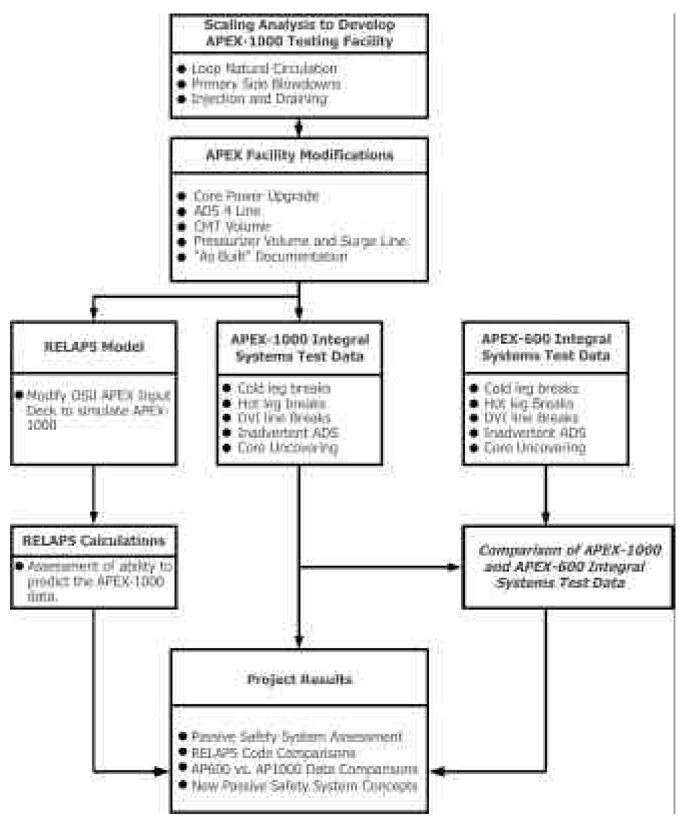


Figure 1. The flow chart shows the sequence of events in the OSU AP1000 Test and Analysis Program.

All of the system components will be constructed of stainless steel and will be capable of consistent operation at 400 psia while at saturation temperatures. All of the primary system components will be insulated to minimize heat loss.

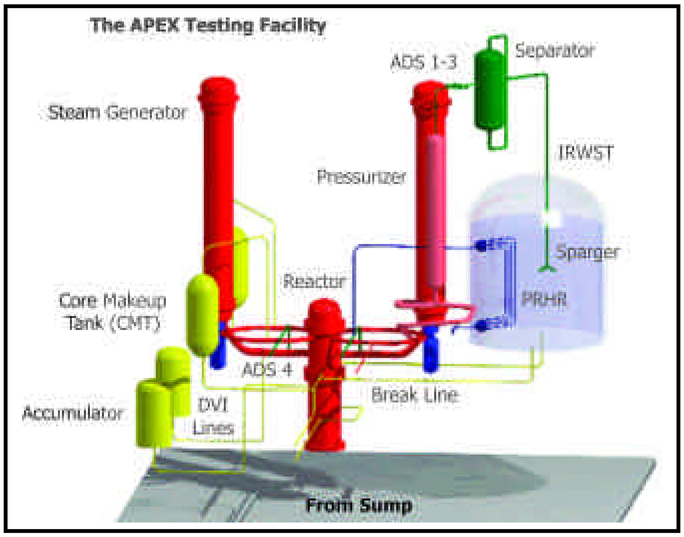


Figure 2. The figure is a graphical representation of the OSU APEX-1000 Testing Facility.

Planned Activities

Having completed the scaling analysis, the key modifications will be made to the APEX Testing Facility and "as-built" drawings will be developed. All of the re-configured facility documentation will be labeled as APEX-1000. The facility modification details will also be incorporated into the APEX RELAP5 input deck in order to compare thermal hydraulic computer codes to the APEX-1000 test data. A variety of pre-test and post-test calculations will be performed.

After modifying the APEX Testing Facility, a wide range of integral systems tests will be performed. This includes cold leg breaks, hot leg breaks, inadvertent ADS operation, direct vessel injection (DVI) line breaks, and core uncovering tests. The APEX-1000 test results will be compared to the APEX-600 test results to gain an understanding of the effects of high power on passive safety system performance. All of the results will be documented in a final report. This includes a complete description of the test results, key comparisons to APEX-600 data, RELAP5 comparisons to key APEX-1000 data, and suggestions for new passive safety system concepts for next-generation, high-power reactors.

NUCLEAR ENERGY RESEARCH INITIATIVE

Engineering and Physics Optimization of Breed & Burn Fast Reactor Systems

Primary Investigator: Michael J. Driscoll, Massachusetts Institute of Technology

Collaborators: Idaho National Engineering & Environmental Laboratory; Argonne National Laboratory

Project Number: 02-005

Project Start Date: September 2002

Project End Date: September 2005

The term "breed & burn" (B&B) is used to refer to fast reactors in which reload fuel has a significantly lower enrichment than that required to sustain criticality. The deficit is made up by breeding new fissile material in the fresh fuel faster than its depletion in the older fuel in the core. In the ideal case, only depleted uranium make-up is required in the steady state, and no reprocessing is required. The most compelling attribute of B&B systems is their virtual guarantee of sustainability via significantly better utilization of uranium resources. The concept is found in the NERI field of endeavor F-1: "Nuclear Engineering-Advanced Nuclear Energy Systems," where Generation IV goals are addressed and summarized in Table 1. B&B is not an entirely new concept, although only a handful of investigators have published on this topic since it was first mentioned in 1958 by the Russian physicist, Feinberg. Moreover, the emphasis to date has been almost entirely on reactor physics. Complicated fuel shuffling schemes have been suggested, including temporary introduction of a moderator, and fuel in-core residence times in excess of 100 years (!) were put forward uncritically in this prior work.

The present proposal will determine the feasibility of achieving significant B&B benefits in practical plant designs. These will likely be based on gas coolant and carbide or metal fuel, to avoid introducing too much moderation, which spoils the ultra-hard spectra needed by this concept. High power density is also needed to accelerate fuel throughput and move quickly to an equilibrium fuel cycle. Overall project coordination will be provided by MIT; INEEL and ANL-WEST will contribute significantly in their areas of special expertise and in collaborative work on shared tasks.

NUCLEAR ENERGY RESEARCH INITIATIVE

Evaluation of Integral Pressurizers for Generation IV PWR Concepts

Primary Investigator: David K. Felde, Oak Ridge National Laboratory

Collaborators: Westinghouse Electric Company

Project Number: 02-018

Project Start Date: September 2002

Project End Date: September 2005

Integral pressurizers are a key design feature of several proposed pressurized water reactor (PWR) designs included in the Generation IV reactor concepts based on light water reactor (LWR) technology. These Integrated Primary System Reactor (IPSR) concepts are characterized by the inclusion of the entire primary system within a single pressure vessel, including the steam generators and pressurizer. For higher power output, forced circulation is required and designs include internal pumps, although relatively large natural-to-forced circulation flow ratios remain as favorable operating characteristics. As in conventional PWRs, the pressurizer and its interaction with the primary system are important factors in the dynamic control and stability of the reactor during both normal and off-normal conditions.

This project will develop the tools and methods needed to evaluate and characterize the functionality of integral pressurizer designs for new Generation IV reactor concepts. As part of this process, it is the intent to map the performance of integral pressurizers as a function of their design parameters (e.g., gas content, vapor volume, interface with the primary system). Based on the detailed analysis of pressurizer performance, the work will aim to propose pressurizer design solutions that will allow the reactor system to be simplified or withstand more severe accident scenarios.

The integral reactors are expected to have significant multi-dimensional effects because the entire primary circuit is accommodated within the reactor vessel. Analyses will be performed using a 3-D system thermal hydraulic code (RELAP5-3D) to characterize the inflow/outflow surge rates and the operating envelope required for the pressurizer of a generic integral reactor. As part of this process, safety objectives will be defined and incorporated into the models and applicable bounding conditions. Selection will be made of initiating events leading to the highest degree of reactor over-pressurization or de-pressurization. In parallel, characteristics of the supporting systems (e.g., spray, heaters) will be studied and operating regimes defined. The functional design requirements determined in this phase of the study will be used as input to more detailed analysis of specific integral pressurizer characteristics, as described below.

For each pressurizer type (e.g. steam, steam-gas), the design details that allow it to meet the functional requirements will be identified. The major issues affecting the design of an internal pressurizer for integral reactors are related to the behavior of the interface between the primary circuit and the pressurizer. For a steam pressurizer, the interface dictates the thermal stratification and thereby the separation of the steam phase from the generally cooler primary circuit flow. The steam-gas pressurizer relies on the addition of inert gas to the steam phase to reach the operating pressure conditions. The interface has to be designed properly to avoid large gas concentrations in the primary coolant. In order to examine the thermal and gas transport characteristics of the interface and to characterize the response of the pressurizer to system pressure changes, a comprehensive analysis using a Computational Fluid Dynamics (CFD) code will be made.

Complementary models, correlations, or calculational techniques will be applied, as necessary, to the base CFD code structure using the same solution algorithm. The following principal processes, occurring in the pressurizer, will be analyzed in order to evaluate a particular pressurizer design: expansion and associated compression of the gas phase; spray condensation (condensation on falling droplets); internal condensation and water flashing; condensate fall rate; bubble rise rate; diffusion controlled mass transfer on phase boundaries; gas solubility and content in water under different conditions; and so forth.

A small experiment will be undertaken to refine the CFD model and validate the code results. A scaled model of the pressurizer section of an integral reactor will be built. In-surge and out-surge flows will be determined on the basis of expected operational regimes developed previously. Steam and steam-gas combinations will be tested, varying the gas content. Different interface designs for the pressurizer will be investigated in order to minimize the mass diffusion and maximize the impulse transport through the interface.

Having established characteristics of the specific pressurizer types, an investigation will be made of factors that could potentially affect system stability, and in particular, their effect on the natural circulation capabilities of the reactor. For the gas-steam pressurizers in particular, the effects associated with the non-condensable gas content and its compressibility will be addressed. With information from the CFD studies, an effort will be made to evaluate the circulation stability of a generic natural circulation system typical of the integral PWR designs. The study will parametrically assess the effect of pressurizer parameters (e.g., gas content, gas volume, interface areas) on natural circulation stability.

The systematic approach developed in the previous phases of the study will be applied to a down-selected reactor/pressurizer design. Models and tools developed for the generic reactor will be applied to the specific design. Transient system code runs will be carried out to analyze the evolution of the same initiating events analyzed for the generic reactor for verification of results and of compliance with the established safety and design criteria. In addition, the stability tests and the analysis of the relationship between pressurizer parameters and natural circulation stability performed for the generic reactor will be repeated for the specific reactor/pressurizer design.

From the results of this study, it will be possible to develop a sound approach to pressurizer selection as a function of reactor characteristics, giving the designer the ability to make detailed design decisions. Critical design details and characteristics will be identified and guidelines for application of the results will be developed and provided in the final report.

The proposed project is a collaboration between ORNL and the Westinghouse Electric Company, with non-funded international support from the Politecnico di Milano (POLIMI), Italy, and the Comissao Nacional de Energia Nuclear (CNEN), Brazil.

NUCLEAR ENERGY RESEARCH INITIATIVE

Nuclear-Energy-Assisted Plasma Technology for Producing Hydrogen

Primary Investigator: Peter C. Kong, Idaho National Engineering and Environmental Laboratory

Project Number: 02-030

Project Start Date: September 2002

Project End Date: September 2005

Most of the energy currently used in the world comes from fossil energy sources. The world's supply of fossil energy is finite and presents a variety of environmental problems from mining and extraction activities to air pollution caused by emissions when they are burned. Although the world's store of fossil fuels has already diminished, the world's demand for energy will not diminish. Moreover, the developed world is less willing to tolerate environmental damage from future methods of energy production, and is actively seeking new solutions for its energy needs. Among all the alternative energy possibilities, hydrogen is the strongest candidate to meet the global demand for energy without sacrificing the environment-it does not emit any air pollutants. Since hydrogen's energy density is high, it is a highly efficient energy source that can be used for transportation, heating, and power generation. If it can be produced, transported, and stored economically and cleanly in large quantities, hydrogen can replace fossil fuels in

- Automobiles and other personal transportation,
- Industrial processes, and
- Distributed power applications.

Several technologies exist to produce hydrogen, but they have disadvantages. For example, fossil fuel reformers produce hydrogen from methane, gasoline, natural gas, or other fossil fuels. These reformer systems are complex and capital intensive and the hydrogen produced is of low purity. The technologies also create polluting emissions from the carbon, sulfur, and nitrogen compounds inherent in the fossil fuel. Additionally, hydrogen from reformers contains carbon monoxide, which requires separation to produce pure hydrogen. Hydrogen generated from fossil fuels must still be stored—either compressed in cylinders or liquefied and stored as a cryogenic liquid. Both of these storage mechanisms have limited consumer appeal, particularly for transportation and residential power applications.

Thermal cracking of fossil energy sources also produces hydrogen and solid carbon residue, but the processes require separation to obtain pure hydrogen. Electrolysis is also used to generate hydrogen from water. The water-electrolysis process consumes significant amount of electricity with low conversion efficiency, and is designed only for stationary use. Another option is the use of metal hydrides, but only for storing the energy that hydrogen produces. Metal hydride systems still require a source of hydrogen gas for producing hydrogen fuel. An improved process to produce hydrogen must be developed.

Sodium borohydride is a safe and concentrated hydrogen carrier compound and can store an impressive amount of hydrogen. For example, 1 liter of 44-weight percent sodium borohydride solution at 1 atmosphere can release about 130 grams of hydrogen. Sodium borohydride releases more hydrogen and has a higher density of hydrogen than other sources of hydrogen. For example, cryogenic liquefied hydrogen has a density of 70 gm/lit. Hydrogen pressurized to 6,000 psi has a density of only 36 gm/lit. Rare-earth-nickel alloys can store hydrogen up to a density slightly higher than liquid hydrogen but still quite a bit less than that of sodium borohydride. However, the alloy is very expensive and not as easily handled as a liquid. The borohydride solution is also much easier and safer to handle than liquid or high-pressure hydrogen. The current gasoline-distribution infrastructure for automobiles can be easily converted to dispense "sodium borohydride fuel" for vehicles.

Sodium borohydride can be produced from sodium borate although at present, no technology exists to do so economically. Development of a nuclear-power-assisted plasma technology is proposed for economically mass-producing sodium borohydride from sodium borate.

A successful nuclear-power-assisted plasma technology to convert sodium borate to sodium borohydride will have a long-term significant economical benefit to the nuclear power industry. During peak operation, nuclear power reactors will generate electricity to meet peak commercial demand, and during off-peak operation, the nuclear reactor will supply electricity and nuclear process heat to produce sodium borohydride. Producing sodium borohydride during off-peak hours will in turn increase the demand for the nuclear industry.

We have assembled a team of highly experienced and qualified researchers to develop a new nuclear-energy-assisted plasma technology to produce hydrogen. Our proposed technology does not have the disadvantages of existing hydrogen-producing technologies. In contrast to the existing hydrogen-producing technologies, the proposed process to mass-produce sodium borohydride from sodium borate is

- Efficient,
- Economical,
- Environmentally acceptable, and
- Safe.

It will also help the country convert to a hydrogen economy. Sodium borohydride solution is also much easier and safer to handle than liquid or high-pressure hydrogen. This technology will help in facilitating hydrogen as the energy source to power the world.

NUCLEAR ENERGY RESEARCH INITIATIVE

Coupling of High Temperature, Lead-Cooled, Closed Fuel Cycle Fast Reactors to Advanced Energy Converters

Primary Investigator: James J. Sienicki, Argonne National Laboratory (ANL)

Collaborators: Oregon State University; Forschungszentrum Karlsruhe

Project Number: 02-065

Project Start Date: September 2002

Project End Date: September 2005

The three-year program of research and development aims to develop high-temperature modular nuclear plant concepts that take advantage of the sustainability benefits of a fast neutron spectrum core; the safety benefits of molten lead primary coolant; and the cost advantages of modular construction, factory fabrication, and simplification with natural circulation heat transport. At the same time these plant concepts would achieve sufficiently high coolant temperatures to drive an advanced power conversion system-for example, a gas turbine Brayton cycle using supercritical carbon dioxide-providing efficiencies competitive with those claimed for high temperature gas reactor (HTGR) concepts. Unlike HTGRs, the concepts provide the sustainability and economic fuel cycle benefits of a liquid metal-cooled fast reactor. Unlike conventional fast reactors, utilization of a gas turbine, compressors, recuperator, precooler, intercoolers, and supporting components offers radical plant simplification, reduced staffing levels, and cost savings, as well as greater efficiency relative to a Rankine cycle water-steam system, but at traditional liquid metal reactor (LMR) temperatures of approximately 550°C.

Lead (T_{mp} = 327°C) is selected as the primary coolant based upon its high boiling temperature (1,740°C) and inertness; lead does not burn when exposed to air. When operating at higher-temperature, Brayton-cycle conditions, the high melting point (327ºC) ceases to be as large a problem as under Rankine cycle conditions. Lead is less corrosive than bismuth, especially at elevated temperatures. Small module power (e.g., approximately 400 MWth) enables 100+ percent natural circulation of the primary coolant, enhancing plant simplification, reliability, cost savings, and passive safety. The fast spectrum core with negative reactivity feedbacks facilitates nearly autonomous operation whereby the core power automatically adjusts itself to load changes as a result of inherent physical processes. Heat rejection to a gas also favors autonomous load following over a wider range of power levels. The reactivity feedback coefficients together with a passive reactor exterior cooling system utilizing air driven by natural circulation also effect passive core power shutdown in the event of accidents such as a loss-of-heat sink.

The utilization of supercritical carbon dioxide as the Brayton cycle working fluid could provide cycle efficiencies of about 45 percent at core outlet temperatures as low as 550°C. Increases in efficiency to well over 50 percent could be achieved with supercritical carbon dioxide by increasing the lead temperature. The achievement of such high efficiencies, even at traditional fast reactor temperatures, is a result of the low amount of work required to compress carbon dioxide immediately above the critical pressure as compared to the case of nonsupercritical He or CO_2. The supercritical CO_2 approach is particularly attractive because it works at temperatures traditionally reached in LMR systems, whereas if gaseous (i.e., nonsupercritical) helium or carbon dioxide were utilized as the Brayton cycle working fluid, cycle efficiencies of 45 to 50 percent would be achieved only by heating the gas to a core outlet temperature of nearly 900°C. In that case, a main challenge would be the identification of cladding, fuel, and structural materials for use with molten lead at elevated temperatures as well as innovative techniques for the manufacture of components from these materials. The need for experiemental data to undertake further development will be evaluated.

The goal to move away from the Rankine steam cycle to modern energy converters will not stop at the Brayton cycle. As an alternate to a gas-turbine energy converter, magnetohydrodynamic (MHD) generators for direct power conversion can potentially take advantage of the high electrical conductivity of liquid metals such as molten lead. A MHD generator converts fluid kinetic energy to electrical energy.

Argonne National Laboratory, as well as the Forschungszentrum Karlsruhe in Karlsruhe, Germany, will partner with Oregon State University (OSU) on various aspects of the project.

NUCLEAR ENERGY RESEARCH INITIATIVE

Experimental Verification of Magnetic Insulation for Direct Energy Conversion Fission Reactors

Primary Investigator: Donald B. King, Sandia National Laboratories

Collaborators: Texas A&M University; General Atomics

Project Number: 02-068

Project Start Date: September 2002

Project End Date: September 2005

This project will investigate the feasibility of direct energy conversion by designing, building, and testing prototype cells with advanced magnetic isolation and insulation technologies.

This research will establish the feasibility of developing reactors that directly capture the energy of nuclear fission fragments to produce electricity. With no intermediate conversion to thermal energy, the efficiencies of such reactors are not subject to classical thermodynamic limitations. The potential maximum efficiency of a direct energy conversion reactor is approximately 50 percent and is independent of temperature. As high temperatures and pressures are not required, large safety margins and passively safe design should be achievable. These advantages, combined with integral power conversion and modular design, present an opportunity to develop a low-cost reactor system.

Concepts to achieve direct energy conversion of fission fragments were investigated during the 1950s and 1960s. Experiments demonstrated the basic physics of the concept, but technical shortfalls prevented the attainment of operational goals. Magnetic solenoids required to capture the fission fragments and insulators to withstand high voltage gradients due to capture did not exist. Dramatic improvements in relevant technological disciplines have occurred since this time, including magnetic and insulation development, and are directly applicable to this direct energy conversion project.

NUCLEAR ENERGY RESEARCH INITIATIVE

Innovative Low-Cost Approaches to Automating QA/QC of Fuel Particle Production Using On-line Nondestructive Methods for Higher Reliability

Primary Investigator: Ronald L. Hockey, Pacific Northwest National Laboratory

Collaborators: General Atomics; Iowa State University; Oak Ridge National Laboratory

Project Number: 02-103

Project Start Date: September 2002

Project End Date: September 2005

Sub-millimeter TRISO fuel particles having multiple layers of pyrocarbon and silicon carbide are used in several current research systems and proposed advanced nuclear reactor fuel designs. The performance of these micro-spheres is a key component in system containment and depends particularly heavily on the properties and performance of the silicon carbide layer. Present day quality assurance and quality control (QA/QC) methods, done manually and in many cases destructively, are unable to test economically the large numbers of fuel particles that are required in fuel fabrication for advanced reactor concepts.

The project is designed to provide the United States with key enabling advanced inspection technologies for application to "fuel particles" (TRISO particles). These technologies are required for the economical production of reactor fuels being proposed for several Nuclear Power 2010 and Generation IV designs. The project will explore, adapt, develop, and demonstrate innovative nondestructive test methods that will provide in-line measurements for qualification of multilayered (TRISO) nuclear fuel particles and provide improved QA/QC.

The project will focus on nondestructive technologies that can be automated for production speeds, particularly those with potential for implementation as in-line measurements. Supporting studies will be performed on techniques with potential for either on-process implementation (where average properties of a batch of particles can be characterized) and those that can be used to give enhanced off-line measurements. The primary task for both the in-line and off-line tests will be to provide standard signatures for both acceptable particles and the most problematic types of defects. The data from the signatures will be used as the basis for establishing and demonstrating a multiple attribute "Quality Index," which can be used to integrate data and can be applied to grade both individual and batches of particles.

The primary thrust of the project will be in-line measurements, and this will focus on the assessment of the potential of electrical property measurements, which have potential for noncontact, rapid, volumetric property determination. It is proposed that these will be combined with advanced optical measurements to give shape and size assessment. It is intended that the data from these two methodologies will be integrated to give a Quality Index for both individual particles and batches of particles.

Supporting studies and particle characterization will be performed using high resolution computed tomography, acoustic microscopy, and resonance ultrasonic spectroscopy. Data from these studies will be used to assess the performance of the integrated electrical and optical measurements. If required to provide additional in-line characterization, an additional technology, from among those identified above, will be developed to support the optical and electrical measurements. The potential for the use of low-frequency ultrasonics as an on-process tool for monitoring batch properties will be evaluated.

The benefits from the successful completion of the project follow:

- Demonstration of the feasibility of using electrical measurements (eddy current/dielectric constant), integrated with advanced optical testing, for on-line TRISO QA/QC
- Development and demonstration of a Quality Index to measure both individual and batch conformity
- Sets of well-characterized surrogate particles for use in QA/QC technology evaluations

- An assessment of the capabilities of using acoustic microscopy, resonant ultrasound spectroscopy (RUS) and high resolution computed tomography for both in-line and/or advanced off-line NDE/QA-QC measurements on TRISO fuel particles
- Provision of proof-of-principle data for the use of transmission and diffuse field ultrasound for on-process monitoring and to provide improved quality control
- Availability of a family of QA/QC tools, which have been evaluated on surrogate fuel particles and are ready to be transitioned for use on a pilot plant scale for a TRISO fuel, or similar nuclear fuel fabrication line

NUCLEAR ENERGY RESEARCH INITIATIVE

Model Based Transient Control and Component Degradation Monitoring in Generation IV Nuclear Power Plants

Primary Investigator: James Holloway, University of Michigan

Collaborators: Westinghouse Electric Company; Sandia National Laboratory; Dominion Generation

Project Number: 02-113

Project Start Date: September 2002

Project End Date: September 2005

A project is proposed to support the development of advanced nuclear power technology and to help position it as a highly competitive and safe method of energy generation. The project will develop a highly advanced and integrated methodology for constructing model-based control systems for Generation IV-based nuclear generating stations. The project will also develop an advanced approach for monitoring nuclear plant systems for system degradations. These two tasks are united by their reliance on smart sensor networks that map sensor signals to plant state information. This plant state information is used to connect models of plant state to the actual plant state. Nonlinear state-space control algorithms based on a Hamiltonian formulation of the control problem can then provide robust and automatic plant control in a wide variety of plant transient maneuvers, such as start-up, shutdown, and load follow maneuvers, including large or total load rejections. By providing smooth transient control without reactor trip these control systems can greatly improve both plant safety and economics. The quest for long-life cores in highly integrated and modular reactor designs places great demands on the already difficult maintenance systems of nuclear power stations. Development is proposed of a systematic statistical methodology for monitoring plant performance degradation. By solving a Master equation for the probability of finding the plant in a given system state and having a given set of component states, it is possible to determine the probability that the plant is in a given component state, given a set of plant sensor signals. Such advanced degradation monitoring will allow nuclear plant operators to optimize plant maintenance procedures that are subject to both safety and economic factors.

The work proposed will provide the nuclear engineering community with two new capabilities:

(1) A method to develop robust nonlinear control algorithms that combine plant sensor measurements with a physical model of key plant systems.

(2) A methodology for plant system degradation monitoring based on comparing plant sensor readings with a physical model of key plant systems.

These methods for fusing sensor data with physical models of plant systems will allow nuclear plant engineers to design optimal maintenance and control strategies at the onset for the new generation of nuclear plants. They will provide nuclear plant operators with tools to operate their plants safely and efficiently within the complex energy market of the 21st century.

NUCLEAR ENERGY RESEARCH INITIATIVE

Centralized Hydrogen Production from Nuclear Power: Infrastructure Analysis and Test-Case Design Study

Primary Investigator: William A. Summers, Savannah River Technology Center

Collaborators: General Atomics; University of South Carolina; Entergy Nuclear, Inc.

Project Number: 02-160

Project Start Date: September 2002

Project End Date: September 2005

The National Energy Plan proposed by President Bush foresees that energy from hydrogen will have an increasing role in the national economy, and that a large-scale, hydrogen-based, energy economy will augment the current fossil fuel energy economy, reducing the nation's dependence on imported petroleum.

Achieving the vision of a hydrogen-based economy requires safe, cost-effective methods of producing and distributing hydrogen in the quantities needed to support a major part of the Nation's energy and transportation needs. Also required is a strategy for making the transition to such an economy, especially because the infrastructure needs for centralized hydrogen production are considerably more complex than those for the alternative method of distributed hydrogen production using electrolysis.

The objective of this research is to identify, characterize, and evaluate the critical technical and economic issues associated with a new and innovative approach to centralized hydrogen production-thermochemical decomposition of water using heat from a nuclear reactor. These issues include hydrogen production, storage, distribution, and end-user integration.

Outcomes of this research will include the information needed to evaluate the technical feasibility and economic attractiveness of nuclear reactor-produced hydrogen, as well as the detailed characteristics for a commercial prototype system and an analysis of the economics of building such a system. The resulting methodology will help in developing the actual production facilities and related infrastructure. Technology gaps identified in this study will be the basis for future research projects designed to overcome barriers to implementation.

This research will define the process and infrastructure needed for nuclear hydrogen production to become a reality. In the process, the project will take advantage of many past and current studies of either production or infrastructure issues. However, although hydrogen infrastructure studies have been performed, no comprehensive analysis exists of an integrated nuclear reactor-thermochemical production system and the required supporting infrastructure. Therefore, this study will be unique in examining the integration of nuclear and thermochemical processes with infrastructure, and will define the engineering and economic factors needed to deliver nuclear-produced hydrogen to end users.
This study will build on existing design studies being supported by the Department of Energy through the Nuclear Energy Research Initiative, and will have two phases:

- Phase A, Infrastructure Analysis-conduct a comprehensive examination of nuclear hydrogen production methods and related hydrogen infrastructure systems. From this phase will come the detailed information needed to evaluate the technical feasibility and economic attractiveness of nuclear-thermochemical hydrogen production and to build a prototype commercial system.
- Phase B, Test Case Preconceptual Design-develop more specific, detailed results by conducting a test-case design study and economic analysis that hypothesizes thermochemical hydrogen production at a specific site and with a specific end-user. The Department of Energy's Savannah River Site and an existing local chemical plant will be the basis for this study. This phase is particularly applicable to determining the feasibility of using nuclear reactor-produced hydrogen for industrial applications such as oil refineries and chemical plants during transition to a large-scale, hydrogen-based energy economy.

The proposed project is led by the Westinghouse Savannah River Company (WSRC) through its applied research and development laboratory, Savannah River Technology Center (SRTC). Supporting SRTC will be a highly qualified team of two major industrial partners, General Atomics (GA) and Entergy Nuclear, Inc.; a leading university partner, the University of South Carolina (USC); and two experienced hydrogen consultants, Mr. Robert B. Moore, (retired, Air Products and Chemicals), and Dr. Joan Ogden, Princeton University. The team members will contribute extensive expertise and experience in every discipline required to make this project a success.

WSRC, with over 50 years of nuclear and hydrogen expertise as well as extensive research and development (R&D), production, and project experience, will take the lead in defining the hydrogen storage, transmission, and delivery systems, and will provide overall project management. GA brings a wealth of experience in nuclear reactor design and the thermochemical process, and will lead the process definition in these two areas. Entergy Nuclear, the nation's second largest nuclear power plant operator, will validate the preliminary design and cost information and provide an overall utility-company perspective on the project. USC has considerable expertise in hydrogen and fuel cell technology, and will lead the effort to develop an economic model to evaluate the various hydrogen infrastructure scenarios. USC will be supported in their tasks by the two key consultants, Robert Moore and Dr. Joan Ogden, both of whom have extensive backgrounds in hydrogen infrastructure studies and planning.

Hydrogen produced from nuclear power not only has many attractive environmental advantages, including the reduced emissions of nitrous oxides, sulfur, and global warming gases. It also has the potential to impact the Nation's energy security by reducing a dependence on imported oil. However, uncertainty about the supply system and the price of end-use hydrogen precludes an accurate assessment of hydrogen's potential future contribution to the national energy supply. This unique, comprehensive study will help define that future by providing valuable information for assessing the merits of nuclear hydrogen production, both for near-term hydrogen supply for chemical plants and for longer-term hydrogen supply for the hydrogen economy.

NUCLEAR ENERGY RESEARCH INITIATIVE

Near-Core and In-Core Neutron Radiation Monitors for Real Time Neutron Flux Monitoring and Reactor Power Level Measurements

Primary Investigator: Douglas S. McGregor, Kansas State University

Project Number: 02-174

Project Start Date: September 2002

Project End Date: September 2005

There is a need for neutron radiation detectors capable of withstanding intense radiation fields, of performing "near-core" reactor measurements, of pulse mode and current mode operation, and of discriminating neutron signals from background gamma ray signals. The detectors should also be tiny enough to be inserted directly into a nuclear reactor without significantly reducing or altering the neutron flux. Such devices can be used to monitor nuclear reactor power levels in "real-time."

A method is proposed here to accomplish these requirements with a new type of compact neutron detector fabricated through the utilization of present day micro-machining technology. The basic device consists of a miniaturized gas-filled chamber with either ^{10}B or ^{235}U inside coatings. The device width can be reduced to 1 mm or less while retaining up to 7 percent thermal neutron detection efficiency. The device is extremely radiation-hard and should continue to operate after exposure to neutron fluences exceeding 10^{16} n/cm^2. Furthermore, the compact design reduces background gamma ray interference. The device can be manufactured from a variety of materials, including common semiconductor and insulating materials. Overall, the device will be inexpensive to reproduce and operate.

The compact devices will be deployed in and around the KSU TRIGA reactor and tested as real-time neutron flux and power monitors. Inversion models will be developed to correlate the detector measurements with reactor power levels and performance.

NUCLEAR ENERGY RESEARCH INITIATIVE

Development of a Supercritical Carbon Dioxide Brayton Cycle: Improving PBR Efficiency and Testing Material Compatibility

Primary Investigator: Chang H. Oh, Idaho National Engineering and Environmental Laboratory (INEEL)

Project Number: 02-190

Project Start Date: September 2002

Project End Date: September 2005

Generation IV reactors will need to be intrinsically safe, having a proliferation-resistant fuel cycle and several advantages relative to existing light water reactor (LWR) systems. They, however, must still overcome certain technical issues and the cost barrier before it can be built anywhere in the world. South Africa wants to build the German-type PBR, but it has no detailed calculations on postulated accidents using the German design, there are unresolved technical issues, and the cost factor is still unfavorable.

A fundamental study is proposed to determine how to make the gas-cooled reactors safer and more economical, to meet the world's power requirements for the next generation.

The proposed project establishes a nuclear power cost goal of 3.3 cents/kWh in order to compete with fossil combined-cycle, gas-turbine power generation. This goal requires approximately a 30 percent reduction in power cost for state-of-the-art nuclear plants. It has been demonstrated that this large cost differential can be overcome only by technology improvements that lead to a combination of better efficiency and more compatible reactor materials. The proposal outlines

(1) development of a supercritical carbon dioxide Brayton cycle,

(2) improvement of the plant net efficiency by using the supercritical carbon dioxide Brayton cycle, and

(3) testing of material compatibility at supercritical conditions and high temperatures.

Developing a Supercritical Carbon Dioxide Brayton Cycle and Improving Efficiency: Supercritical carbon dioxide (SC CO_2) has a moderate critical constant, T_c = 31°C and P_c = 7.29 MPa, and has a unique heat transfer capacity at the supercritical condition. For example, the density of SC CO_2 is higher than helium by a factor 2000 and higher than supercritical water (SCW) by a factor of 1.5 to 2 at a temperature of 920°C and above. SC CO_2 has advantages over even SCW and helium in terms of heat transfer and higher molecular weight. The heat capacity term (mass flow rate times heat capacity) is higher than that for helium because of higher density. SCW has a much higher pressure (22 MPa) at the critical condition and a very narrow range of high heat capacity around the critical temperature of 374°C.

INEEL calculations for the Brayton cycle indicate that SC CO_2 has a 55 percent cycle efficiency versus 41 percent for helium for the reference PBR design of INEEL and MIT. The higher efficiency is achieved at a lower turbine inlet temperature for SC CO_2, 535.4°C versus 850°C for helium. The higher molecular weight results in less work in compression, which contributes to a higher efficiency for the SC CO_2 Brayton cycle.

The use of SC CO_2 as a coolant in the secondary PBR is very attractive because the core outlet temperature can be increased, which will increase the plant net efficiency by more than 60 percent.

Testing Material Compatibility: It is proposed to characterize the creep deformation of Inconel MA 754 and 758 over a range of temperatures from 850°C to 1050°C and stresses within the power law creep regime. By varying the temperature at constant stress and the stress at constant temperature, it will be possible to determine the numerical values of the activation energy and power law exponent for creep, and whether a threshold stress formalism applies for these materials or if the ARZT type model is more appropriate. This characterization will allow the proper constitutive equation to be determined, so that the deformation behavior can be calculated for a long service time in the temperature and stresses expected for the advanced reactor concept described above.

In addition to suitable mechanical properties, the alloys must resist environmental degradation for extended periods of time for the conditions expected in this reactor concept. Preliminary analysis suggests that the nickel-based alloys with 20 to 30 percent Cr content will exhibit reasonable resistance to degradation by supercritical CO_2. An investigation is proposed of the interaction of MA 754 and 758 in supercritical CO_2 using thermogravimetric analysis combined with surface analysis to examine the possible chemical interaction mechanism(s) (e.g., breakdown of the passivating Cr oxide or carburization), at temperatures and pressures of interest.

Nuclear Energy Research Initiative

Hydrogen Production Plant Using the Modular Helium Reactor

Primary Investigator: Arkal Shenoy, General Atomics

Collaborators: Idaho National Engineering & Environmental Laboratory; Entergy Nuclear Inc.; Texas A&M University

Project Number: 02-196

Project Start Date: September 2002

Project End Date: September 2005

There is a large and growing demand for hydrogen both in the United States and worldwide, with the bulk of the hydrogen being produced by steam reforming of methane. Hydrogen, along with electricity, are expected to dominate the world energy system in the long term. As the United States and the rest of the world transitions to a hydrogen economy, hydrogen will be used increasingly by the transportation, residential, industrial, and commercial sectors of the energy market. Eventually, an alternative source of hydrogen will be needed because

(1) the demand for natural gas is outpacing its production, and

(2) steam reforming of natural gas is not environmentally friendly because it produces the greenhouse gas CO_2.

A promising alternative source of hydrogen is to use process heat from a high-temperature nuclear reactor to drive a set of chemical reactions that produce hydrogen. Preliminary evaluations have shown that the sulfur-iodine (SI) process can produce hydrogen with high efficiency when driven by the 850°C to 950°C process heat from a Modular Helium Reactor (MHR). The SI process produces highly pure H_2 and O_2, with formation, decomposition, regeneration, and recycle of the reagents H_2SO_4 and HI. Preliminary economic assessments have shown that an MHR-driven SI plant can produce hydrogen economically, especially if the cost of natural gas increases because of increased demand. The MHR's high-temperature capability, advanced stage of development relative to other high-temperature reactor concepts, and passive-safety features make it ideally suited as the heat source for producing hydrogen. The work proposed here is the next logical step-to develop a conceptual design for a hydrogen production plant that integrates an MHR reactor system with an SI-cycle hydrogen production plant. Figure 1 shows an artist's conception of the integrated plant, referred to as the H_2-MHR. As an added measure of safety, the reactor system is located below grade and isolated from the H_2 production system through the use of intermediate heat exchangers. The H_2-MHR represents a significant advancement of nuclear technology and offers a safe and potentially economical source of clean, renewable hydrogen.

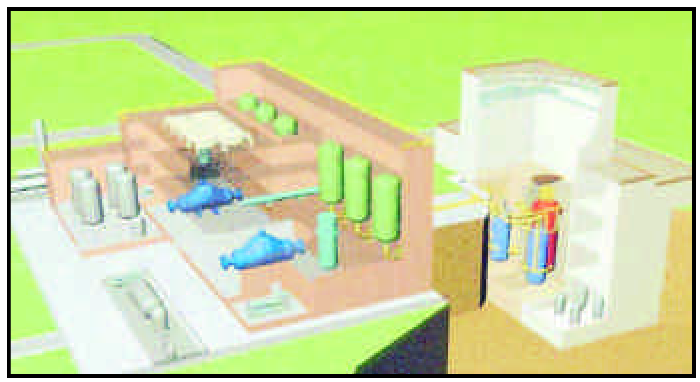


Figure 1. The schematic of the H_2-MHR plant shows the modular helium reactor integrated with an H_2 production plant (source: Generation IV Higher Temperature Reactor Materials Workshop, La Jolla, CA, March 18, 2002; figure prepared by Japan Atomic Energy Research Institute).

The proposed project will span a three-year period. During the first six months of the project, a systems-engineering approach will be used to prepare a Plant Functions and Requirements document. This document will provide the basis for developing the conceptual design of the H_2-MHR plant. Annual reports will be issued at the end of Project Years 1 and 2 to document the work performed during these years. An H_2-MHR Conceptual Design Report will be the final deliverable, to be issued at the end of Project Year 3. The work proposed here supports all of the NERI program objectives and will provide the Department of Energy, utilities, and energy-policy planners with precisely the type of information needed to make decisions regarding additional research and development for producing hydrogen using nuclear energy.

A team consisting of General Atomics (GA), Idaho National Engineering and Environmental Laboratory (INEEL), Entergy, and Texas A&M has been assembled to perform the proposed work. GA will be the lead organization and will be responsible for project management, plant definition, reactor system design, and plant integration. Texas A&M will have lead responsibility for developing the hydrogen production system design. INEEL will have lead responsibility for performing plant assessments, trade studies, and sensitivity analyses. Entergy, a major nuclear utility with a strong interest in hydrogen production, will function as a non-funded participant and will periodically review the design work from the perspective of a potential customer. Each organization is highly qualified and highly motivated to work on this project.

Nuclear Energy Research Initiative

Nuclear Reactor Power Monitoring Using Silicon Carbide Semiconductor Radiation Detectors

Primary Investigator: Don Miller, Ohio State University

Collaborators: Westinghouse Savannah River Company; General Atomics

Project Number: 02-207

Project Start Date: September 2002

Project End Date: September 2005

This proposal is directed to the design, development, and assessment of a new paradigm in ex-core neutron flux monitoring for nuclear power plants. The proposed system is based on the use of silicon carbide (SiC) neutron sensors configured in arrays, technology that has been under development by Westinghouse since 1994. There are two fundamental characteristics of such arrays that distinguish them from current technology, which employs a variety of gas-filled neutron sensors:

(1) They operate in pulse mode over a wide dynamic range, which permits pulse spectroscopy, and

(2) they are relatively small in physical size, which permits measurements at discrete physical locations.

To access these characteristics, a collaborative program among the Ohio State University (OSU), Westinghouse, and General Atomics (GA) is proposed that will investigate the use of SiC-based sensor arrays as ex-core neutron monitors in the International Reactor Innovative and Secure Reactor (IRIS), which is being developed by Westinghouse Electric Company and in the prismatic-core, gas turbine modular helium-cooled reactor (GT-MHR), which is being developed by General Atomics.

The proposed three year research program will identify advantages and disadvantages associated with the use of SiC neutron sensors, examine solutions for overcoming difficulties associated with their use and will develop and evaluate methods for improving the performance of SiC based neutron sensor channels.

The following deliverables are identified as key products of the proposed research program.

(1) Selection of the optimum locations for SiC-based neutron power monitors in both the IRIS and the GT-MHR. Factors that will be considered include the power monitoring requirements as well as expected detector sensitivity and presence of gamma ray background.

(2) Evaluation of other applications and opportunities offered by SiC-based neutron power monitors that will include but not be limited to prospects for on-line fault identification and diagnosis using pulse height and pulse shape analysis and the use of miniature SiC detectors to define axial, azimuthal, and radial flux profiles.

(3) A prototype SiC-based, neutron-power monitor with high event rate electronics whose performance will be evaluated in the Ohio State University Research Reactor under neutron fluence rate conditions that provide pulse rates that are commensurate with monitoring requirements in both the IRIS and the GT-MHR.